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

December 3, 2018

Mr. Mano Nazar
President, Nuclear Division
and Chief Nuclear Officer
Florida Power & Light Company
Mail Stop EX/JB
700 Universe Blvd.
Juno Beach, FL 33408

SUBJECT: TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4 - ISSUANCE OF AMENDMENTS REGARDING ADOPTION OF RISK-INFORMED COMPLETION TIMES IN TECHNICAL SPECIFICATIONS (CAC NOS. MF5455 AND MF5456; EPID L-2014-LLA-0002)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 284 to Renewed Facility Operating License No. DPR-31 and Amendment No. 278 to Renewed Facility Operating License No. DPR-41 for the Turkey Point Nuclear Generating Unit Nos. 3 and 4, respectively. The amendments revise the Technical Specifications in response to the application from Florida Power & Light Company dated December 23, 2014, as supplemented by letters dated June 16, and August 11, 2016; February 9, April 27, and October 30, 2017; and February 15, March 22, June 12, and September 6, 2018.

The amendments revise various Technical Specifications to permit the use of risk-informed completion times for selected required actions. The NRC staff's safety evaluation of the amendments is enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Michael J. Wentzel, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosures:
1. Amendment No. 284 to DPR-31
2. Amendment No. 278 to DPR-41
3. Safety Evaluation

cc: Listserv

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 284
Renewed License No. DPR-31

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

   A. The application for amendment by Florida Power & Light Company (the licensee) dated December 23, 2014, as supplemented by letters dated June 16, and August 11, 2016; February 9, April 27, and October 30, 2017; and February 15, March 22, June 12, and September 6, 2018; 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 Renewed Facility Operating License and Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Renewed Facility Operating License No. DPR-31 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 284, are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 180 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: December 3, 2018

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR GENERATING UNIT NO. 4

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 278
Renewed License No. DPR-41

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

   A. The application for amendment by Florida Power & Light Company (the licensee) dated December 23, 2014, as supplemented by letters dated June 16, and August 11, 2016; February 9, April 27, and October 30, 2017; and February 15, March 22, June 12, and September 6, 2018; 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.

Enclosure 2

2. Accordingly, the license is amended by changes to the Renewed Facility Operating License and Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Renewed Facility Operating License No. DPR-41 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 278, are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 180 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: December 3, 2018

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 284 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 278 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-41

TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

Replace pages 3 and 8 of Renewed Facility Operating License No. DPR-31 with the attached pages 3 and 8. The revised pages are identified by amendment number and contain a marginal line indicating the area of change.

Replace pages 3 and 8 of Renewed Facility Operating License No. DPR-41 with the attached pages 3 and 8. The revised pages are identified by amendment number and contain a marginal line indicating the area of change.

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

Remove	Insert	Remove	Insert
3/4 3-5	3/4 3-5	3/4 7-10	3/4 7-10
3/4 3-6	3/4 3-6	3/4 7-14	3/4 7-14
3/4 3-14	3/4 3-14	3/4 7-16	3/4 7-16
3/4 3-15	3/4 3-15	3/4 8-2	3/4 8-2
3/4 3-19	3/4 3-19	3/4 8-4	3/4 8-4
3/4 3-22A	3/4 3-22A	3/4 8-13	3/4 8-13
3/4 5-3	3/4 5-3	3/4 8-14	3/4 8-14
3/4 5-4	3/4 5-4	3/4 8-19	3/4 8-19
3/4 6-3	3/4 6-3	3/4 8-20	3/4 8-20
3/4 6-12	3/4 6-12	3/4 8-21	3/4 8-21
3/4 6-14	3/4 6-14	3/4 8-22	3/4 8-22
3/4 6-16	3/4 6-16	6-14A	6-14A
3/4 7-3	3/4 7-3	--	6-14B

E. Pursuant to the Act and 10 CFR Parts 40 and 70 to receive, possess, and use at any time 100 milligrams each of any source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactively contaminated apparatus;

F. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Turkey Point Units Nos. 3 and 4.

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

A. Maximum Power Level

The applicant is authorized to operate the facility at reactor core power levels not in excess of 2644 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 284, are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Final Safety Analysis Report

The licensee's Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on November 1, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than July 19, 2012.

The Final Safety Analysis Report supplement as revised on November 1, 2001, described above, shall be included in the next scheduled update to the Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following the issuance of this renewed license. Until that update is complete, the licensee may make changes to the programs described in such supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

H. PAD TCD Safety Analyses

1. PAD 4.0 TCD has been specifically approved for use for the Turkey Point licensing basis analyses. Upon NRC's approval of a revised generic version of PAD that accounts for Thermal Conductivity Degradation (TCD), FPL will within six months:

   a. Demonstrate that PAD 4.0 TCD remains conservatively bounding in licensing basis analyses when compared to the new generically approved version of PAD w/TCD, or
   b. Provide a schedule for the re-analysis using the new generically approved version of PAD w/TCD for any of the affected licensing basis analyses

I. FPL is authorized to implement the Risk Informed Completion Time Program as approved in License Amendment No. 284 subject to the following conditions:

1. FPL will complete the items listed in the table of implementation items in the enclosure to FPL letter L-2018-118 dated June 12, 2018 prior to implementation of the Risk Informed Completion Time Program.

2. The risk assessment approach and methods, shall be acceptable to the NRC, be based on the as-built, as-operated, and maintained plant, and reflect the operating experience of the plant as specified in RG 1.200. Methods to assess the risk from extending the completion times must be PRA methods accepted as part of this license amendment, or other methods approved by the NRC for generic use. If the licensee wishes to change its methods, and the change is outside the bounds of this license condition, the licensee will seek prior NRC approval via a license amendment.

4. This renewed license is effective as of the date of issuance, and shall expire at midnight July 19, 2032.

FOR THE NUCLEAR REGULATORY COMMISSION

Signed by
Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:
Appendix A – Technical Specifications for Unit 3
Appendix B – Environmental Protection Plan

Date of Issuance: June 6, 2002

E. Pursuant to the Act and 10 CFR Parts 40 and 70 to receive, possess, and use at any time 100 milligrams each of any source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactively contaminated apparatus;

F. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Turkey Point Units Nos. 3 and 4.

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

A. Maximum Power Level

The applicant is authorized to operate the facility at reactor core power levels not in excess of 2644 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 278, are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Final Safety Analysis Report

The licensee's Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on November 1, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than April 10, 2013.

The Final Safety Analysis Report supplement as revised on November 1, 2001, described above, shall be included in the next scheduled update to the Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following the issuance of this renewed license. Until that update is complete, the licensee may make changes to the programs described in such supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

H. PAD TCD Safety Analyses

1. PAD 4.0 TCD has been specifically approved for use for the Turkey Point licensing basis analyses. Upon NRC's approval of a revised generic version of PAD that accounts for Thermal Conductivity Degradation (TCD), FPL will within six months:

    a. Demonstrate that PAD 4.0 TCD remains conservatively bounding in licensing basis analyses when compared to the new generically approved version of PAD w/TCD, or
    b. Provide a schedule for the re-analysis using the new generically approved version of PAD w/TCD for any of the affected licensing basis analyses

I. FPL is authorized to implement the Risk Informed Completion Time Program as approved in License Amendment No. 278 subject to the following conditions:

1. FPL will complete the items listed in the table of implementation items in the enclosure to FPL letter L-2018-118 dated June 12, 2018 prior to implementation of the Risk Informed Completion Time Program.

2. The risk assessment approach and methods, shall be acceptable to the NRC, be based on the as-built, as-operated, and maintained plant, and reflect the operating experience of the plant as specified in RG 1.200. Methods to assess the risk from extending the completion times must be PRA methods accepted as part of this license amendment, or other methods approved by the NRC for generic use. If the licensee wishes to change its methods, and the change is outside the bounds of this license condition, the licensee will seek prior NRC approval via a license amendment.

4. This renewed license is effective as of the date of issuance, and shall expire at midnight April 10, 2033.

FOR THE NUCLEAR REGULATORY COMMISSION

Signed by
Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:
Appendix A – Technical Specifications for Unit 4
Appendix B – Environmental Protection Plan

Date of Issuance: June 6, 2002

TABLE 3.3-1 (Continued)

TABLE NOTATION

*	When the Reactor Trip System breakers are in the closed position and the Control Rod Drive System is capable of rod withdrawal.
**	When the Reactor Trip System breakers are in the open position, one or both of the backup NIS instrumentation channels may be used to satisfy this requirement. For backup NIS testing requirements, see Specification 3/4.3.3.3, ACCIDENT MONITORING.
***	Reactor Coolant Pump breaker A is tripped by underfrequency sensor UF-3A1(UF-4A1) or UF-3B1(UF-4B1). Reactor Coolant Pump breakers B and C are tripped by underfrequency sensor UF-3A2(UF-4A2) or UF-3B2(UF-4B2).
#	Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.
##	Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than 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 HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored per Specification 4.2.4.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes and verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

ACTION 10- With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable, except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection					
a. Manual Initiation	2	1	2	1 2, 3, 4	27
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1 2, 3, 4	14
c. Containment Pressure - High	3	2	2	1 2, 3	26
d. Pressurizer Pressure - Low	3	2	2	1 2, 3#	26
e. High Differential Pressure Between the Steam Line Header and any Steam Line	3/steam line	2/steam line in any steam line	2/steam line	1 2, 3#	26

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Line flow--High Coincident with:	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3*	26
Steam Generator Pressure--Low	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2. 3*	26
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3*	25
2. Containment Spray					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Containment Pressure--High-High Coincident with:	3	2	2	1, 2, 3	26
Containment Pressure--High	3	2	2	1, 2, 3	26
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	27
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

TABLE 3.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater### (Continued)					
b. Stm. Gen. Water Level-- Low-Low	3/steam generator	2/steam generator in any steam generator	2/steam generator	1, 2, 3	26
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
d. Bus Stripping	1/bus	1/bus	1/bus	1, 2, 3	23(a)
e. Trip of all Main Feed-water Pumps Breakers	1/breaker	(1/breaker) /operating pump	(1/breaker) /operating pump	1, 2	23(b)
7. Loss of Power					
a. 4.16 kV Busses A and B (Loss of Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4	18
b. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Undervoltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18
Coincident with: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-2 (Continued)

TABLE NOTATION (Continued)

ACTION 24A - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 7 days restore the inoperable channel to OPERABLE status or place the Control Room Emergency Ventilation System in the recirculation mode.

ACTION 24B - With the number of OPERABLE channels two less than the Minimum Channels OPERABLE requirement, either:

1. Immediately place the Control Room Emergency Ventilation System in the recirculation mode with BOTH Control Room emergency recirculation fans operating, OR

2. a. Immediately place the Control Room Emergency Ventilation System in the recirculation mode with ONE Control Room emergency recirculating fan operating, AND

   b. Restore at least one inoperable channel to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both Units simultaneously, then be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 25 - With number of OPERABLE channels one less than the Total number of channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours or in accordance with the Risk Informed Completion Time Program. For subsequent required DIGITAL CHANNEL OPERATIONAL TESTS the inoperable channel may be placed in bypass status for up to 4 hours.

ACTION 26 - With one channel inoperable, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST or TRIP ACTUATING DEVICE OPERATIONAL TEST provided the inoperable channel is place in the tripped condition within 6 hours or in accordance with the Risk Informed Completion Time Program.

ACTION 27 - With one channel inoperable, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 The following Emergency Core Cooling System (ECCS) equipment and flow paths shall be OPERABLE:

a. Four Safety Injection (SI) pumps, each capable of being powered from its associated OPERABLE diesel generator#, with discharge flow paths aligned to the RCS cold legs,*

b. Two RHR heat exchangers,

c. Two RHR pumps with discharge flow paths aligned to the RCS cold legs,

d. A flow path capable of taking suction from the refueling water storage tank as defined in Specification 3.5.4, and

e. Two flow paths capable of taking suction from the containment sump.

APPLICABILITY: MODES 1, 2, and 3**

ACTION:

a. With one of the following components inoperable:

   1. RHR heat exchanger,
   2. RHR suction flow path from the containment sump,
   3. RHR parallel injection flow path, or
   4. SI parallel injection flow path

   Restore the inoperable component to OPERABLE status within 72 hours, or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. Deleted

c. With one of the four required Safety Injection pumps or its associated discharge flow path inoperable and the opposite unit in MODE 1, 2, or 3, restore the pump or flow path to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.***

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*Only three Safety Injection (SI) pumps (two associated with the unit and one from the opposite unit), each capable of being powered from its associated OPERABLE diesel generator#, with discharge flow paths aligned to the RCS cold leg are required if the opposite unit is in MODE 4, 5, or 6.

**The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the Safety Injection flow paths isolated pursuant to Specification 3.4.9.3 provided that the Safety Injection flow paths are restored to OPERABLE status prior to Tavg exceeding 380°F. Safety Injection flow paths may be isolated when Tavg is less than 380°F.

***The provisions of Specification 4.0.4 are not applicable.

#Inoperability of the required diesel generators does not constitute inoperability of the associated Safety Injection pumps.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

d. With two of the four required Safety Injection pumps or their associated discharge flow paths inoperable and the opposite unit in MODE 1, 2, or 3, restore one of the two inoperable pumps or flow paths 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 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

e. With one of the three required Safety Injection pumps or its associated discharge flow path inoperable and the opposite unit in MODE 4, 5, or 6, restore the pump or flow path 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.

f. With a required Safety Injection pump OPERABLE, but not capable of being powered from its associated diesel generator, restore the capability within 14 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.

g. With an ECCS subsystem inoperable due to an RHR pump or its associated discharge flow path being inoperable, restore the inoperable RHR pump or its associated discharge flow path 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.

h. With the suction flow path from the refueling water storage tank inoperable, restore the suction flow path to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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, or during the performance of containment air lock surveillance and/or testing requirements, 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 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.

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 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.

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 RWST and manually transferring suction to the containment sump via the RHR System.

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

ACTION:

a. With one Containment Spray System inoperable restore the inoperable Spray 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 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 in COLD SHUTDOWN within the following 30 hours. Restore both Spray Systems to OPERABLE status within 72 hours of initial loss 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

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. 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 in its correct position* and that power is available to flow path components that require power for operation;

b. By verifying that each Containment Spay pump's developed head at the test flow point is greater than or equal to the required developed head, when tested in accordance with the INSERVICE TESTING PROGRAM.

c. In accordance with the Surveillance Frequency Control Program by verifying containment spray locations susceptible to gas accumulation are sufficiently filled with water.

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* Not required to be met for system vent flow paths opened under administrative control.

CONTAINMENT SYSTEMS

EMERGENCY CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Three emergency containment cooling units shall be OPERABLE.

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

a. With one of the above required emergency containment cooling units inoperable restore the inoperable cooling unit 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 two or more of the above required emergency containment cooling units inoperable, restore at least two cooling units to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore all of the above required cooling units to OPERABLE status within 72 hours of initial loss 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

4.6.2.2 Each emergency containment cooling unit shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by starting each cooler unit from the control room and verifying that each unit motor reaches the nominal operating current for the test conditions and operates for at least 15 minutes.

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

   1) Verifying that two emergency containment cooling units start automatically on a safety injection (SI) test signal, and

   2) Verifying a cooling water flow rate of greater than or equal to 2000 gpm to each cooler.

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each containment isolation valve shall be OPERABLE with isolation times less than or equal to required isolation times.

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

ACTION:

*With one or more isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

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

b. 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 containment isolation valve secured in the isolation position, or

c. 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, or

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

SURVEILLANCE REQUIREMENTS

4.6.4.1 The isolation valves 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.

*CAUTION: The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent auxiliary feedwater trains including 3 steam supply flowpaths, 3 pumps and associated discharge water flowpaths shall be OPERABLE.[(1)(2)]

APPLICABILITY: MODES 1, 2 and 3

ACTION:

NOTE: LCO 3.0.4.b is not applicable to the required auxiliary feedwater trains when entering Mode 1.

1) With one of the two required independent auxiliary feedwater trains inoperable, either restore the inoperable train to an OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program, or place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours.

2) With both required auxiliary feedwater trains inoperable, within 2 hours either restore both trains to an OPERABLE status, or restore one train to an OPERABLE status and follow ACTION statement 1 above for the other train. If neither train can be restored to an OPERABLE status within 2 hours, verify that both Standby Feedwater Pumps are capable of providing makeup flow to the steam generators and place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours. Otherwise, initiate corrective action to restore at least one auxiliary feedwater train to an OPERABLE status as soon as possible and follow ACTION statement 1 above for the other train.

3) With a single auxiliary feedwater pump inoperable, within 4 hours, verify OPERABILITY of two independent auxiliary feedwater trains, or follow ACTION statements 1 or 2 above as applicable. Upon verification of the OPERABILITY of two independent auxiliary feedwater trains, restore the inoperable auxiliary feedwater pump to an OPERABLE status within 30 days, or place the operating unit(s) in at least HOT STANDBY within 6 hours* and in HOT SHUTDOWN within the following 6 hours.

4) With a single steam supply flowpath inoperable, within 4 hours verify OPERABILITY of two independent steam supply flowpaths or follow ACTION statement 1 or 2 above as applicable. Upon verification of the OPERABILITY of two independent steam supply flowpaths, restore the inoperable steam supply flowpath to OPERABLE status within 7 days of discovery, or place the affected Unit(s) in at least HOT STANDBY within 6 hours* and in HOT SHUTDOWN within the following 6 hours.

NOTES:

(1) One steam supply flowpath shall be OPERABLE in each AFW train and the third steam supply flowpath (via MOV-3-1404 for Unit 3 and MOV-4-1404 for Unit 4) shall be OPERABLE and aligned to either AFW train but not both simultaneously.

(2) During single and two unit operation, one pump shall be OPERABLE in each train and the third auxiliary feedwater pump shall be OPERABLE and capable of being powered from, and supplying water to either train, except as noted in ACTION 3 of Technical Specification 3.7.1.2. The third auxiliary feedwater pump (normally the "C" pump) can be aligned to either train to restore OPERABILITY in the event one of the required pumps is inoperable.

*If this ACTION applies to both units simultaneously, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 24 hours; otherwise be in MODE 2 within the next 6 hours.

MODES 2 and 3:

With one or more MSIV inoperable, subsequent operation in MODE 2 or 3 may continue provided:

1. The inoperable MSIVs are closed within 8 hours, and

2. The inoperable MSIVs are verified closed once per 7 days.

Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested in accordance with the INSERVICE TESTING PROGRAM. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

3/4.7.2 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 The Component Cooling Water System (CCW) shall be OPERABLE with:

a. Three CCW pumps, and

b. Two CCW heat exchangers.

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

ACTION:

a. With only two CCW pumps with independent power supplies OPERABLE, restore the inoperable CCW pump to OPERABLE status within 30 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With only one CCW pump OPERABLE or with two CCW pumps OPERABLE but not from independent power supplies, restore two pumps from independent power supplies 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.

c. With less than two CCW heat exchangers OPERABLE, restore two heat exchangers to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.2 The Component Cooling Water System (CCW) shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program, by verifying that two heat exchangers and one pump are capable of removing design basis heat loads.

3/4.7.3 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The Intake Cooling Water System (ICW) shall be OPERABLE with:

a. Three ICW pumps, and
b. Two ICW headers.

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

ACTION:

a. With only two ICW pumps with independent power supplies OPERABLE, restore the inoperable ICW pump to OPERABLE status within 14 days 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 only one ICW pump OPERABLE or with two ICW pumps OPERABLE, but not from independent power supplies, restore two pumps from independent power supplies 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.

c. With only one ICW header OPERABLE, restore two headers 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.

SURVEILLANCE REQUIREMENTS

4.7.3 The Intake Cooling Water System (ICW) 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; and

b. In accordance with the Surveillance Frequency Control Program during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a SI test signal, and

2) Each Intake Cooling Water System pump starts automatically on a SI test signal.

3) Interlocks required for system operability are OPERABLE.

LIMITING CONDITION FOR OPERATION (Continued)

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

ACTION:

NOTE: LCO 3.0.4.b is not applicable to diesel generators.

a. With one of two startup transformers or an associated circuit inoperable, demonstrate the OPERABILITY of the other startup transformer and its associated circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the inoperable startup transformer is the associated startup transformer and became inoperable while the unit is in MODE 1, reduce THERMAL POWER to ≤30% RATED THERMAL POWER within 24 hours, or restore the inoperable startup transformer and associated circuits to OPERABLE status within the next 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. If THERMAL POWER is reduced to ≤30% RATED THERMAL POWER within 24 hours or if the inoperable startup transformer is associated with the opposite unit restore the startup transformer and its associated circuits to OPERABLE status within 30 days of the loss of OPERABILITY, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. If the inoperable startup transformer is the associated startup transformer, and became inoperable while the unit was in MODE 2, 3, or 4, restore the startup transformer and its associated 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 6 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

b. With one of the required diesel generators inoperable, demonstrate the OPERABILITY of the above required startup transformers and their associated circuits by performing Surveillance Requirement 4.8.1.1.1.a 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 preventative maintenance or testing, demonstrate the OPERABILITY of the remaining required diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours, unless the absence of any potential common mode failure for the remaining diesel generators is determined. If testing of remaining required diesel generators is required, this testing must be performed regardless of when the inoperable diesel generator is restored to OPERABILITY. Restore the inoperable diesel generator to OPERABLE status within 14 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 COLD SHUTDOWN within the following 30 hours.

c. With one startup transformer and one of the required diesel generators inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a on the remaining

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** 72 hours if inoperability is associated with Action Statement 3.8.1.1.c.

ACTION (Continued)

SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously. With only one startup transformer and associated circuits restored, perform Surveillance Requirement 4.8.1.1.1a on the OPERABLE Startup transformer at least once per 8 hours, and restore the other startup transformer and its associated circuits to OPERABLE status or shutdown in accordance with the provisions of Action Statement 3.8.1.1a with time requirements of that Action Statement based on the time of initial loss of a startup transformer. This ACTION applies to both units simultaneously.

f. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two startup transformers and their associated circuits by performing the requirements of Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least 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. Restore all required diesel generators to OPERABLE status within 14 days from time of initial loss 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.

g. Following the addition of the new fuel oil* to the Diesel Fuel Oil Storage Tanks, with one or more diesel generators with new fuel oil properties outside the required Diesel Fuel Oil Testing Program limits, restore the stored fuel oil properties to within the required limits within 30 days.

h. With one or more diesel generators with stored fuel oil total particulates outside the required Diesel Fuel Oil Testing Program limits, restore the fuel oil total particulates to within the required limits within 7 days.

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* The properties of API Gravity, specific gravity or an absolute specific gravity; kinematic viscosity; clear and bright appearance; and flash point shall be confirmed to be within the Diesel Fuel Oil Testing Program limits, prior to the addition of the new fuel oil to the Diesel Fuel Oil Storage Tanks.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following D.C. electrical sources shall be OPERABLE:*#

a. 125-volt D.C. Battery Bank 3A or spare battery bank D-52 and associated full capacity charger(s)
   1) 3A1 powered by motor control center (MCC) 3C with EDG 3A OPERABLE, or
   2) 3A2 powered by MCC 4D with EDG 4A and 4B OPERABLE, or
   3) 3A1 powered by MCC 3C with EDG 3A OPERABLE and 3A2 powered by MCC 4D with EDG 4A and 4B OPERABLE,

b. 125-volt D.C. Battery Bank 3B or spare battery bank D-52 and associated full capacity charger(s)
   1) 3B1 powered by MCC 3B with EDG 3B OPERABLE, or
   2) 3B2 powered by MCC 4D with EDG 4A and 4B OPERABLE, or
   3) 3B1 powered by MCC 3B with EDG 3B OPERABLE and 3B2 powered by MCC 4D with EDG 4A and 4B OPERABLE,

c. 125-volt D.C. Battery Bank 4A or spare battery bank D-52 and associated full capacity charger(s)
   1) 4A1 powered by MCC 4C with EDG 4A OPERABLE, or
   2) 4A2 powered by MCC 3D with EDG 3A and 3B OPERABLE, or
   3) 4A1 powered by MCC 4C with EDG 4A OPERABLE and 4A2 powered by MCC 3D with EDG 3A and 3B OPERABLE,

d. 125-volt D.C. Battery Bank 4B or spare battery bank D-52 and associated full capacity charger(s)
   1) 4B1 powered by MCC 4B with EDG 4B OPERABLE, or
   2) 4B2 powered by MCC 3D with EDG 3A and 3B OPERABLE, or
   3) 4B1 powered by MCC 4B with EDG 4B OPERABLE and 4B2 powered by MCC 3D with EDG 3A and 3B OPERABLE.

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

ACTION:

a. With one or more of the required battery chargers OPERABLE but not capable of being powered from its associated OPERABLE diesel generator(s), restore the capability within 72 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

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* All battery chargers required to satisfy the LCO shall be powered from separate MCCs.

\# Inoperability of the required EDG's specified in the LCO requirements below does not constitute inoperability of the associated battery chargers or battery banks.

D.C. SOURCES

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

b. With one of the required battery banks inoperable, or with none of the full-capacity chargers associated with a battery bank OPERABLE, restore all battery banks to OPERABLE status and at least one charger associated with each battery bank to OPERABLE status within two hours* or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and its associated full capacity charger(s) 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 129 volts on float charge and the battery charger(s) output voltage is ≥ 129 volts, and

3) If two battery chargers are connected to the battery bank, verify each battery charger is supplying a minimum of 10 amperes, or demonstrate that the battery charger supplying less than 10 amperes will accept and supply the D.C. bus load independent of its associated battery charger.

b. In accordance with the Surveillance Frequency Control Program and within 7 days after a battery discharge with battery terminal voltage below 105 volts (108.6 volts for spare battery D-52), or battery overcharge with battery terminal voltage above 143 volts, by verifying that:

1) The parameters in Table 4.8-2 meet the Category B limits,

2) The average electrolyte temperature of every sixth cell is above 60°F, and

3) There is no visible corrosion at either terminals or connectors, or verify battery connection resistance is:

Battery	Connection	Limit (Micro-Ohms)
3B, 4A	inter-cell / termination inter-cell (brace locations) transition cables or total battery connections	≤ 29 ≤ 30 ≤ 125 ≤ 1958
Battery	**Connection**	**Limit (Micro-Ohms)**
3A, 4B, D-52	inter-cell / termination inter-cell (brace locations) transition cables or total battery connections	≤ 35 ≤ 40 ≤ 125 ≤ 2463

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

1) The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

*Can be extended to 24 hours if the opposite unit is in MODE 5 or 6 and each of the remaining required battery chargers is capable of being powered from its associated diesel generator(s).

j. 120 Volt AC Vital Panel 3P09 and 3P24 energized from its associated inverter connected to D.C. Bus 4A.****

k. 120 Volt AC Vital Panel 4P09 and 4P24 energized from its associated inverter connected to D.C. Bus 4A.****

l. 125 Volt D.C. Bus 3D01 energized from an associated battery charger and from Battery Bank 3A or spare battery bank D-52,

m. 125 Volt D.C. Bus 3D23 energized from an associated battery charger and from Battery Bank 3B or spare battery bank D-52,

n. 125 Volt D.C. Bus 4D01 energized from an associated battery charger and from Battery Bank 4B or spare battery bank D-52, and

o. 125 Volt D.C. Bus 4D23 energized from an associated battery charger and from Battery Bank 4A or spare battery bank D-52

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

ACTION:

a. With one of the required trains (3.8.3.1a., b., and c) of A.C. emergency busses not fully energized (except for the required LC's and MCC's associated with the opposite unit), reenergize the train 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 any of the required LC's and/or MCC's associated with the opposite unit inoperable, restore the inoperable LC or MCC to OPERABLE status in accordance with Table 3.8-1 or Table 3.8-2 as applicable or place the unit in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one A.C. vital panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) Reenergize the A.C. vital panel within 2 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital panel from an inverter connected to its associated D.C. bus

****A back-up inverter may be used to replace the normal inverter, provided the normal inverter on the same DC bus for the opposite unit is not replaced at the same time.

ACTION: (Continued)

within 24 hours or in accordance with the Risk Informed Completion Time Program, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

d. With one D.C. bus not energized from its associated battery bank or associated charger, reenergize the D.C. bus from its associated 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 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

SURVEILLANCE REQUIREMENTS

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

* Can be extended to 24 hours if the opposite unit is in MODE 5 or 6 and each of the remaining required battery chargers is capable of being powered from its associated diesel generator(s).

TABLE 3.8-1

APPLICABLE TO UNIT 3 BASED ON UNIT 4 LOAD CENTERS AND MOTOR CONTROL CENTERS INOPERABLE

ALLOWABLE OUTAGE TIMES

Unit 4 Load Centers and Motor Control Centers Inoperable (Any MODE)	Allowable Outage Times (hours) Unit 3 – MODES 1, 2, 3 and 4		
	With AC Trains 3A, 3B, 4A, & 4B OPERABLE	With AC Trains 3A, 3B, & 4A OPERABLE	With AC Trains 3A, 3B, & 4B OPERABLE
LC 4A	N/A	72[a]	N/A
MCC 4A	N/A	N/A	N/A
LC 4C and/or MCC 4C	2*[a]	2*[a]	N/A
LC 4H and/or MCC 4D	2**[a]	2**[a]	2**[a]
LC 4B and/or MCC 4B	2*[a]	N/A	2*[a]
LC 4D	N/A	N/A	72[a]

* If the battery charger powered from the out-of-service LC and/or MCC is not required by LCO 3.8.2.1, the out-of-service time is not applicable (N/A).

** If neither of the battery chargers powered from the out-of-service LC and/or MCC is required by LCO 3.8.2.1, the out-of-service time is 72 hours or in accordance with the Risk Informed Completion Time Program.

a or in accordance with the Risk Informed Completion Time Program

TABLE 3.8-2

APPLICABLE TO UNIT 4 BASED ON UNIT 3 LOAD CENTERS AND MOTOR CONTROL CENTERS INOPERABLE

ALLOWABLE OUTAGE TIMES

Unit 3 Load Centers and Motor Control Centers Inoperable (Any MODE)	Allowable Outage Times (hours) Unit 4 – MODES 1, 2, 3 and 4		
	With AC Trains 4A, 4B, 3A, & 3B OPERABLE	With AC Trains 4A, 4B, & 3A OPERABLE	With AC Trains 4A, 4B, & 3B OPERABLE
LC 3A	N/A	72[a]	N/A
LC 3C and/or MCC 3C	2*[a]	2*[a]	N/A
LC 3H and/or MCC 3D	2**[a]	2**[a]	2**[a]
LC 3B and/or MCC 3B	2*[a]	N/A	2*[a]
LC 3D	N/A	N/A	72[a]

* If the battery charger powered from the out-of-service LC and/or MCC is not required by LCO 3.8.2.1, the out-of-service time is not applicable (N/A).

** If neither of the battery chargers powered from the out-of-service LC and/or MCC is required by LCO 3.8.2.1, the out-of-service time is 72 hours or in accordance with the Risk Informed Completion Time Program.

[a] or in accordance with the Risk Informed Completion Time Program

o. Gas Decay Tank Explosive Gas and Radioactivity Monitoring Program

This Program provides controls for potentially explosive gas mixtures and the quantity of radioactivity contained in the Gas Decay Tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, Postulated Radioactive Release Due to Waste Gas System Leak or Failure.

The Program shall include:

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

2. A surveillance program to ensure that the quantity of radioactivity contained in each Gas Decay Tank is less than the amount that would result in a whole body exposure of 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provision of SR 4.0.2 and SR 4.0.3 are applicable to the Gas Decay Tank Explosive Gas and Radioactivity Monitoring Program surveillance frequencies.

p. 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, "Risk-Informed Technical Specifications Initiative 4b: Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, November 2006. 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 plant configuration change within the scope of the Risk Informed Completion Time Program 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. Use of a RICT is not permitted for entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE.

e. 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.

6.8.5 DELETED

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 284 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 278 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-41

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By application dated December 23, 2014 (Reference 1), as supplemented by letters dated June 16 (Reference 2), and August 11, 2016 (Reference 3); February 9 (Reference 4), April 27 (Reference 5), and October 30, 2017 (Reference 6); and February 15 (Reference 7), March 22 (Reference 8), May 17 (Reference 9), and September 6, 2018 (Reference 10), Florida Power & Light Company (FPL or the licensee) requested changes to the Technical Specifications (TSs) for Turkey Point Nuclear Generating Unit Nos. 3 and 4 (Turkey Point), which are contained in Appendix A of Renewed Facility Operating Licenses DPR-31 and DPR-41. The licensee originally proposed to adopt, with plant-specific variations, Technical Specification Task Force (TSTF) Traveler TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times [CTs] – RITSTF [Risk-Informed TSTF] Initiative 4b" (Reference 11). The U.S. Nuclear Regulatory Commission (NRC) published in the *Federal Register* (FR) a notice of availability of the model safety evaluation for plant-specific adoption of TSTF-505, Revision 1 on March 15, 2012 (77 FR 15399).

On February 24 and 25, 2016, the NRC staff and its contractors from the Pacific Northwest National Laboratory participated in a regulatory audit at the NextEra Energy Offices in Juno Beach, Florida. The 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 June 21, 2016 (Reference 12), the staff issued an audit summary. By electronic mail (e-mail) dated April 14 (Reference 13), April 18 (Reference 14), and June 1, 2016 (Reference 15), the NRC sent the licensee RAIs. By letters dated June 16, and August 11, 2016, the licensee responded to the RAIs.

By letter dated November 15, 2016 (Reference 16), the staff informed the TSTF of its decision to suspend NRC approval of TSTF-505, Revision 1 because of concerns identified during the review of plant-specific license amendment requests for adoption of the traveler. The staff's letter also stated that it would continue reviewing applications already received and site-specific proposals to address the staff's concerns. By letter dated February 9, 2017, the licensee supplemented its application to address the staff's concerns in the letter dated November 15, 2016. By e-mails dated March 30 (Reference 17), and August 10 2017 (Reference 18); March 1 (Reference 19), and May, 17, 2018 (Reference 20), the NRC sent the licensee RAIs. By letters dated April 27, and October 30, 2017; February 15, March 22, and

June 12, 2018, the licensee responded to the RAIs. By letter dated September 6, 2018, the licensee submitted a supplement to its request to account for changes made to certain TSs as a result of the issuance of unrelated amendments.

The licensee's letters dated June 16, and August 11, 2016; February 9, April 27, and October 30, 2017; and February 15, March 22, June 12, and September 6, 2018, provided clarifying information that did not expand the scope of the application and did not change the NRC staff's original proposed no significant hazards consideration (NSHC) determination, as published in the FR on April 28, 2015 (80 FR 23604).

2.0 REGULATORY EVALUATION

2.1. Description of Risk-Informed Completion Times (RICTs) in Technical Specifications

The TSs contain limiting conditions for operation (LCOs), which 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 Turkey Point TSs refer to these remedial actions as ACTIONS, and the ACTIONS typically describe how the LCO can fail to be met. The licensee must take the ACTIONS under designated conditions within specified CTs. Upon expiration of an ACTION's CT, the TSs require the licensee to exit the TSs' operational mode of applicability or follow other prescribed remedial actions, such as shutting down the reactor.

On May 17, 2007 (Reference 21), the NRC staff approved the Nuclear Energy Institute (NEI) report NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines" (Reference 22), subject to the limitations and conditions set forth in the staff's safety evaluation for NEI 06-09. NEI 06-09, Revision 0-A provides a methodology for modifying selected required actions to provide an optional RICT. NEI 06-09, Revision 0-A provides a methodology for extending CTs and, thereby, delay exiting the operational mode of applicability or taking remedial actions if risk is assessed and managed within the limits and programmatic requirements established by a RICT Program or a configuration risk management program. NEI 06-09, Revision 0-A includes the NRC staff's safety evaluation, but does not incorporate the NRC staff positions, limitations, and conditions into the guidance described in the document. Accordingly, NEI 06-09, Revision 0-A is acceptable for referencing by licensees proposing to amend their TSs to implement RMTS when the NRC staff positions, limitations, and conditions described in the NRC staff's safety evaluation dated May 17, 2007, are met.

TSTF-505, Revision 1 provided guidance for requesting license amendments to adopt RICTs in accordance with NEI 06-09, Revision 0-A if the licensee elects to assess and manage risk in accordance with a RICT Program. TSTF-505, Revision 1 proposed the addition of a new program, "Risk-Informed Completion Time Program," to the Administrative Controls section of the TSs that describes the requirements for extending selected CTs and that references NEI 06-09, Revision 0 as the basis for extending the CTs. TSTF-505, Revision 1 proposed new optional CTs for TSs within the scope of the traveler that permit continued operation beyond the existing CTs within the same required action. Use of the new CT requires risk to be assessed, monitored, and managed as measured by the configuration-specific core damage frequency (CDF) and large early release frequency (LERF), using processes and limits specified in NEI 06-09, Revision 0-A. Use of the new CT also requires compensatory measures, or risk management actions (RMAs), and quantitative evaluation of risk sources if probabilistic risk assessment (PRA) models are not available. TSTF-505, Revision 1 also proposed new conditions, required actions, and CTs to address conditions not currently addressed in TSs.

2.2 Licensee's Proposed Changes

The licensee proposed to add a new program, "Risk Informed Completion Time Program," in Section 6.0, "Administrative Controls," of the TSs, which would require adherence to NEI 06-09, Revision 0-A. The proposed new RICT Program would exclude use of a RICT for any configuration that represents a loss of a specified safety function or inoperability of all required trains of a system required to be operable. In its letter dated February 15, 2018, the licensee proposed that the new TS would state:

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, "Risk-Informed Technical Specifications Initiative 4b: Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, November 2006. 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 plant configuration change within the scope of the Risk Informed Completion Time Program 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. Use of a RICT is not permitted for entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE.

e. 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.

The licensee's letter of February 15, 2018, proposed designating the RICT Program as paragraph n in TS 6.8.4. However, by letters dated June 12 and September 11, 2018, the NRC issued license amendments for Turkey Point 3 and 4 that created new programs that were designated as paragraph n and o in TS 6.8.4. As such, the RICT Program will now be designated as paragraph p in TS 6.8.4.

2.2.1 Revision of ACTION Requirements to Incorporate RICT Program

The licensee requested to revise the CTs for the TS required actions in the following list by providing the option to calculate RICTs. In its letter dated February 9, 2017, the licensee provided a complete set of marked-up TS pages that superseded the TS markups previously submitted. This set of requested changes to the TS were subsequently revised in letters dated April 27 and October 30, 2017; and March 22 and September 6, 2018.

<u>TS 3/4.3.1, Reactor Trip System Instrumentation</u>

- Table 3.3-1, Functional Unit 1 ACTION 1 would be revised to allow the option of calculating a RICT to restore the inoperable channel to OPERABLE.
- Table 3.3-1, Functional Unit 19, ACTION 10 would be revised to allow the option of calculating a RICT to restore the inoperable diverse trip feature to OPERABLE.

<u>TS 3/4.3.2, Engineered Safety Features Actuation System (ESFAS) Instrumentation</u>

- Table 3.3-2, Functional Unit 1.a would be revised to apply new ACTION 27, which would state, "With one channel inoperable, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours."
- Table 3.3-2, Functional Unit 1.c would be revised to apply new ACTION 26, which would state, "With one channel inoperable, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST or TRIP ACTUATING DEVICE OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours or in accordance with the Risk Informed Completion Time Program."
- Table 3.3-2, Functional Unit 1.d and 1.e would be revised to apply new ACTION 26.
- Table 3.3-2, Functional Unit 1.f would be revised to apply new ACTION 26 for Steam Line flow--High Coincident and Steam Generator Pressure—Low. Additionally, Functional Unit 1.f would revise ACTION 25 for T_{avg}—Low to allow the option of calculating a RICT to place the inoperable channel in the tripped condition.
- Table 3.3-2, Functional Unit 2.b would be revised to apply new ACTION 26.
- Table 3.3.2, Functional Unit 3.a.(1) would be revised to apply new ACTION 27.
- Table 3.3.2, Functional Unit 6.b would be revised to apply new ACTION 26.

TS 3/4.5.2, ECCS [Emergency Core Cooling System] Subsystems – T_{avg} Greater Than or Equal to 350 degrees Fahrenheit

- TS 3.5.2 - ACTIONS a, d, e, f, g. The ACTIONS in TS 3.5.2 do not specifically address ECCS flow capability. The ACTIONS address inoperability of ECCS components, but operation in accordance with these ACTIONS maintains the minimum assumed ECCS flow capability, which is provided by one residual heat removal (RHR) pump and two safety injection pumps. Therefore, the licensee proposes to apply the RICT Program to TS 3.5.2, ACTIONS a, d, and e. TS 3.5.2, ACTION f addresses the condition of an operable safety injection pump not capable of being powered from its associated emergency diesel generator (EDG). The time provided in ACTION f to restore this capability is the same as the time provided in TS 3.8.1.1 to restore an inoperable EDG to OPERABLE status. The licensee stated that this is a necessary conforming change to permit adopting a RICT in TS 3.8.1.1.

TS 3/4.6.1.3, Containment Air Locks

- TS 3.6.1.3, ACTION b would be revised to allow the option of calculating a RICT to restore the inoperable air lock to OPERABLE.

TS 3/4.6.2.1, Containment Spray System

- TS 3.6.2.1, ACTION a would be revised to allow the option of calculating a RICT to restore the inoperable Containment Spray System to OPERABLE.
- TS 3.6.2.1, ACTION b would be revised to allow the option of calculating a RICT to restore both inoperable Containment Spray Systems to OPERABLE.

TS 3/4.6.2.2, Emergency Containment Cooling System

- TS 3.6.2.2, ACTION a would be revised to allow the option of calculating a RICT to restore the inoperable emergency containment cooling units to OPERABLE.
- TS 3.6.2.2, ACTION b would be revised to allow the option of calculating a RICT to restore all of the required emergency containment cooling units to OPERABLE.

TS 3/4.6.4, Containment Isolation Valves

- TS 3.6.4, ACTION a would be revised to allow the option of calculating a RICT to restore the inoperable valve(s) to OPERABLE.
- TS 3.6.4, ACTIONs b and c would be revised to allow the option of calculating a RICT to isolate the affected penetration.

TS 3/4.7.1.2, Plant Systems – Turbine Cycle – Auxiliary Feedwater System

- TS 3.7.1.2, ACTION 1 would be revised to allow the option of calculating a RICT to restore the inoperable auxiliary feedwater (AFW) train to OPERABLE.

TS 3/4.7.2, Component Cooling Water System

- TS 3.7.2, ACTION b would be revised to allow the option of calculating a RICT to restore two Component Cooling Water (CCW) pumps from independent power supplies to OPERABLE.

TS 3/4.7.3, Intake Cooling Water System

- TS 3.7.3, ACTION a would be revised to allow the option of calculating a RICT to restore the inoperable Intake Cooling Water (ICW) pump to OPERABLE.
- TS 3.7.3, ACTION b would be revised to allow the option of calculating a RICT to restore two ICW pumps from independent power supplies to OPERABLE.

- TS 3.7.3, ACTION c would be revised to allow the option of calculating a RICT to restore two ICW headers to OPERABLE.

TS 3/4.8.1, Electrical Power Systems – Alternating Current Sources - Operating

- TS 3.8.1.1, ACTION a would be revised to allow the option of calculating a RICT to restore the inoperable startup transformer and associated circuits to OPERABLE.
- TS 3.8.1.1, ACTION b would be revised to allow the option of calculating a RICT to restore the inoperable diesel generator to OPERABLE.
- TS 3.8.1.1, ACTION f would be revised to allow the option of calculating a RICT to restore all required diesel generators to OPERABLE.

TS 3/4.8.2, Direct Current Sources - Operating

- TS 3.8.2.1, ACTION a would be revised to allow the option of calculating a RICT to restore the capability to power one or more of the required battery charger from its associated OPERABLE diesel generator(s).
- TS 3.8.2.1, ACTION b would be revised to allow the option of calculating a RICT to restore all battery banks and at least one charger associated with each battery bank to OPERABLE.

TS 3/4.8.3, Onsite Power Distribution - Operating

- TS 3.8.3.1, ACTION a would be revised to allow the option of calculating a RICT to reenergize the train of emergency busses.
- TS 3.8.3.1, ACTION b (Tables 3.8-1 and 3.8-2). The LCO for TS 3.8.3.1 lists the specific electrical buses required to be operable. TS 3.8.3.1 contains ACTION b to provide restoration times for inoperable load centers and motor control centers (MCCs). The licensee proposes to apply the RICT Program to the restoration times contained in Tables 3.8-1 and 3.8-2 associated with ACTION b.
- TS 3.8.3.1, ACTION c would be revised to allow the option of calculating a RICT to reenergize the alternating current (AC) vital panel.
- TS 3.8.3.1, ACTION d would be revised to allow the option of calculating a RICT to reenergize the direct current (DC) bus from its associated battery bank.

2.2.2 Other Proposed TS Changes

The licensee proposed the following additional changes to the TS:

TS 3/4.7.1.5, Main Steam Isolation Valves (MSIVs)
The current ACTION applicable to MODE 1 states:

> With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 24 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours

The licensee proposed to replace “HOT STANDBY” with “MODE 2” and delete the statement, “and in HOT SHUTDOWN within the following 6 hours.”

The licensee also proposed to replace the ACTION statement applicable to MODES 2 and 3, which currently states:

> With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours

The licensee would replace it with the following:

> With one or more MSIVs inoperable, subsequent operation in MODE 2 or 3 may continue provided:
>
> 1. The inoperable MSIVs are closed within 8 hours, and
>
> 2. The inoperable MSIVs are verified closed once per 7 days.
>
> Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

TS 3/4.6.2.1, Containment Spray System
The licensee proposed to delete footnote **, which pertains to an expired one-time CT extension. Footnote ** currently states:

> ** During Unit 3 Cycle 29 only, a one-time extension from 72 hours to 14 days is allowed to perform 3A Containment Spray Pump (3P214A) planned maintenance, provided the following compensatory measures are in place:
>
> - 3B Containment Spray Pump and associated electrical breaker [Guarded]
> - 3A, 3B and 3C Emergency Containment Coolers and associated electrical breakers [Guarded]
> - 3A, 3B Emergency Diesel Generators [Guarded]
> - Unit 3 Startup Transformer and associated onsite AC power distribution system [Guarded]

Typographical Corrections
The licensee also proposed changes to correct typographical errors. In Table 3.3-2, ACTION 24, the licensee proposed to replace "control room" with "Control Room." In TS 3/4.5.2, ACTION g, the licensee proposed to replace "as" with "at." In TS 3.8.2.1, ACTION b, the licensee proposed to replace "oppsite" with "opposite."

2.3 Regulatory Review

The staff considered the following regulatory requirements, policy statements, and guidance during its review of the proposed changes.

Regulatory Requirements

The regulatory requirements related to the content of the TSs are contained in Section 50.36, "Technical Specifications," of Title 10 of the *Code of Federal Regulations* (10 CFR). Section 50.36 of 10 CFR requires TSs to include the following categories related to station operation: (1) safety limits, limiting safety systems settings, and control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports. Section 50.36(c)(2) states, "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met." Section 50.36(c)(5) states, in part, that administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

Section (h)(2) of 10 CFR 50.55a, "Protection and Safety Systems," requires, in part, the protection systems for plants with construction permits issued before January 1, 1971, such as Turkey Point 3 and 4, to be consistent with the plant-specific licensing basis or meet the requirements of Institute of Electrical and Electronics Engineers (IEEE) Standard 603-1991, and the correction sheet dated January 30, 1995. The Turkey Point 3 and 4 Updated Final Safety Analysis Report (UFSAR) (Reference 23), Section 7.2.1, "Design Bases" states that the "reactor protection is designed to meet all presently defined reactor protection criteria and is in accordance with the proposed IEEE 279 'Standard for Nuclear Plant Protection Systems' August 1968." In addition, UFSAR Section 7.2.1 states that "[t]he Eagle 21 instrumentation system is compliant with IEEE 279-1971."

The following clauses from IEEE 279[1] apply to this review:

- Clause 4.2 "Single Failure Criterion" of IEEE 279 requires:

  > Any single failure within the protection system shall not prevent proper protection system action when required.

- Clause 4.11 "Channel Bypass or Removal from Operation" of IEEE 279 requires:

  > The system shall be designed to permit any one channel to be maintained, and when required, tested or calibrated during power operation without initiating a protective action at the systems level. During such operation the active parts of the system shall of themselves continue to meet the single failure criterion.
  >
  > Exception: "One-out-of-two" systems are permitted to violate the single failure criterion during channel bypass provided that acceptable reliability of operation can be otherwise demonstrated.

Section 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" (i.e., the Maintenance Rule), of 10 CFR, requires licensees to monitor the performance or condition of SSCs against licensee established goals in a manner sufficient to provide reasonable assurance that these SSCs are capable of fulfilling their intended functions. Section 50.65(a)(4) requires the assessment and management of the increase in risk that may result from a proposed maintenance activity.

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[1] IEEE 279-1968 and IEEE 279-1971 use the identical language for Clause 4.2 and for Clause 4.11. The NRC staff use the term "IEEE 279" to reference these two clauses in this safety evaluation.

Turkey Point 3 and 4 were licensed prior to the 1971 publication of Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," to 10 CFR Part 50. As such, Turkey Point 3 and 4 are not licensed to the current GDC of 10 CFR Part 50, Appendix A. Section 1.3 of the UFSAR provides a summary of the 1967 GDC proposed by the U.S. Atomic Energy Commission, as amended by the Atomic Industrial Forum (circa October 2, 1967). The NRC staff considered the following proposed GDC as part of its review:

- 1967 Proposed GDC 19 states that protection systems shall be designed for high functional reliability and in-service testability necessary to avoid undue risk to the health and safety of the public.

- 1967 Proposed GDC 20 states that redundancy and independence designed into protection systems shall be sufficient to assure that no single failure on removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served.

- 1967 Proposed GDC 39 states that alternate power systems shall be provided and designed with adequate independency, redundancy, capacity and testability to permit the functioning required of the engineered safety features. Further, 1967 Proposed GDC 39 states that, as a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each system.

As discussed in UFSAR Chapter 8, the licensee performed an evaluation of the site electrical system design in 1982 and concluded that Turkey Point 3 and 4 also comply with Appendix A to 10 CFR Part 50, GDC 17. GDC 17 states that:

> An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Policy Statements

In the Commission's "Final Policy Statement: Technical Specifications for Nuclear Power Plants," dated July 22, 1993 (58 FR 39132), the NRC addressed the use of Probabilistic Safety Analysis (PSA, currently referred to as Probabilistic Risk Assessment or PRA) in STSs [Standard Technical Specifications]. In this 1993 publication, the NRC stated, in part:

> The Commission believes that it would be inappropriate at this time to allow requirements which meet one or more of the first three criteria [of 10 CFR 50.36] to be deleted from Technical Specifications based solely on PSA (Criterion 4). However, if the results of PSA indicate that Technical Specifications can be relaxed or removed, a deterministic review will be performed.

> The Commission Policy in this regard is consistent with its Policy Statement on "Safety Goals for the Operation of Nuclear Power Plants," 51 FR 30028, published on August 21, 1986. The Policy Statement on Safety Goals states in part, "* * * probabilistic results should also be reasonably balanced and supported through use of deterministic arguments. In this way, judgments can be made * * * about the degree of confidence to be given these [probabilistic][2] estimates and assumptions. This is a key part of the process for determining the degree of regulatory conservatism that may be warranted for particular decisions. This defense-in-depth approach is expected to continue to ensure the protection of public health and safety."
>
> The Commission will continue to use PSA, consistent with its policy on Safety Goals, as a tool in evaluating specific line-item improvements to Technical Specifications, new requirements, and industry proposals for risk-based Technical Specification changes.

Approximately 2 years later, the NRC provided additional detail concerning the use of PRA in the "Final Policy Statement: Use of Probabilistic Risk Assessment in Nuclear Regulatory Activities," dated August 16, 1995 (60 FR 42622). In this publication, the Commission stated:

> The Commission believes that an overall policy on the use of PRA methods in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that would promote regulatory stability and efficiency. In addition, the Commission believes that the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach. [...]
>
> PRA addresses a broad spectrum of initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for multiple and common cause failures. The treatment therefore goes beyond the single failure requirements in the deterministic approach. The probabilistic approach to regulation is, therefore, considered an extension and enhancement of traditional regulation by considering risk in a more coherent and complete manner. [...]
>
> Therefore, the Commission believes that an overall policy on the use of PRA in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that promotes regulatory stability and efficiency. This policy statement sets forth the Commission's intention to encourage the use of PRA and to expand the scope of PRA applications in all nuclear regulatory matters to the extent supported by the state-of-the-art in terms of methods and data. [...]

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[2] At 58 FR 39135. Alteration in the original.

Therefore, the Commission adopts the following policy statement regarding the expanded NRC use of PRA:

(1) The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

(2) PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.

(3) PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

(4) The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

Regulatory Guidance

NUREG-1431, Revision 4, Volume 1, "Standard Technical Specifications – Westinghouse Plants" (Reference 24). Although the Turkey Point 3 and 4 TSs are not based on the guidance in NUREG-1431, the STSs provide an acceptable method for licensees of Westinghouse plants to meet the NRC's requirements in 10 CFR 50.36.

Regulatory Guide (RG) 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 25), describes an acceptable risk-informed approach for assessing the nature and impact of proposed permanent licensing basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

RG 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (Reference 26), describes an acceptable risk-informed approach specifically for assessing proposed TS changes. RG 1.177, Revision 1 identifies a three-tiered approach for a licensee's evaluation of the risk associated with a proposed TS CT change, as follows.

- Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174 and RG 1.177. The first tier assesses the impact on plant risk as expressed by the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF). It also evaluates plant risk while equipment covered by the proposed CT is out-of-service, as represented by incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). The limits for ICCDP and ICLERP are consistent with the criteria for incremental core damage probability (ICDP) and incremental large early release probability (ILERP) from the Nuclear Management and Resources Council (NUMARC) 93-01, Revision 4A, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (Reference 27), guidance for managing the risk of on line maintenance activities. ICDP and ILERP are the limits on which the licensee will base the RICT. This guidance was endorsed by the NRC staff in RG 1.160, Revision 3, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (Reference 28), for compliance with the Maintenance Rule, 10 CFR 50.65(a)(4). Tier 1 also addresses PRA quality, including the technical adequacy of the licensee's plant-specific PRA for the subject application. Cumulative risk of the proposed TS change is considered with uncertainty/sensitivity analysis with respect to the assumptions related to the proposed TS change.

- Tier 2 identifies and evaluates any potential risk significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, is removed from service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The purpose of this evaluation is to ensure that there are appropriate restrictions in place such that risk significant plant equipment outage configurations will not occur when equipment associated with the proposed completion time is implemented.

- Tier 3 addresses the licensee's configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and appropriate compensatory measures are taken to avoid risk-significant configurations that may not have been considered when the Tier 2 evaluation was performed. Compared with Tier 2, Tier 3 provides additional coverage to ensure risk-significant plant equipment outage configurations are identified in a timely manner and that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule, which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance testing and corrective and preventive maintenance, subject to the guidance provided in RG 1.177, Section 2.3.7.1 and the adequacy of the licensee's program and PRA model for this application. The CRMP ensures that equipment removed from service prior to or during the proposed extended completion time will be appropriately assessed from a risk perspective.

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 29), describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light-water reactors. RG 1.200 provides regulatory guidance for assessing the technical adequacy of a PRA. RG 1.200, Revision 2 endorses, with clarifications and qualifications, the use of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard, 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" (i.e., the PRA Standard) (Reference 30).

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Chapter 19, Section 19.2, "Review of Risk Information Used to Support Permanent Plant Specific Changes to the Licensing Basis: General Guidance" (Reference 31). Guidance on evaluating PRA technical adequacy is provided in the SRP, Chapter 19, Section 19.1, Revision 3, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed License Amendment Requests after Initial Fuel Load" (Reference 32). More specific guidance related to risk-informed TS changes is provided in SRP, Section 16.1, Revision 1, "Risk-Informed Decisionmaking: Technical Specifications" (Reference 33), which includes changes to TS CTs as part of risk-informed decision making. Section 19.2 of the SRP references the same criteria as RG 1.177, Revision 1, and RG 1.174, Revision 3, and states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption;

2. The proposed change is consistent with the defense-in-depth philosophy;

3. The proposed change maintains sufficient safety margins;

4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement; and

5. The impact of the proposed change should be monitored using performance measurement strategies.

NEI 06-09, Revision 0-A provides a methodology for modifying selected required actions to provide an optional RICT and extending CTs, thereby delay exiting the operational mode of applicability or taking remedial actions if risk is assessed and managed within the limits and programmatic requirements established by a RICT Program or a configuration risk management program. NEI 06-09, Revision 0-A uses processes that are consistent with the requirements of 10 CFR 50.65(a)(4).

Generic Letter (GL) 80-30, "Clarification of the Term 'Operable' as it Applies to Single Failure Criterion for Safety Systems Required by Technical Specifications," provides guidance to licensees that allows for a plant to temporarily depart from the single-failure design criterion when the plant is operating within a TS action requirement.

3.0 TECHNICAL EVALUATION

These proposed amendments provide for the addition of a RICT Program to the Administrative Controls section of the TS and modifies selected Required Action CTs to permit extending the CTs, provided risk is assessed and managed as described in NEI 06-09, Revision 0-A. In accordance with NEI 06-09, Revision 0-A, PRA methods are used to justify each extension to a required action CT based on the specific plant configuration which exists at the time of the applicability of the required action, and are updated when plant conditions change. The licensee's application for the changes in accordance with NEI 06-09, included documentation regarding the technical adequacy of the PRA models used in the CRMP, consistent with the requirements of RG 1.200.

Most TSs identify one or more Conditions for which the LCO may not be met, to permit a licensee to perform required testing, maintenance, or repair activities. Each Condition has an associated required action for restoration of the LCO or for other actions, each with some fixed time interval, referred to as the CT, which identifies the time interval permitted to complete the required action. Upon expiration of the CT, the licensee is required to shut down the reactor or follow the required action(s) stated in the ACTIONS requirements. 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 within specified limits and programmatic requirements. The specified safety function or performance level of TS required equipment is unchanged, and the required action(s), including the requirement to shut down the reactor are also unchanged; only the CTs for the required actions are extended by the RICT Program.

3.1 Review of Key Principles

Revision 1 of RG 1.177 and RG 1.174, Revision 3, identify five key safety principles to be applied to risk-informed changes to the TSs. Each of these principles is addressed in NEI 06-09, Revision 0-A. The NRC staff's evaluation of the licensee's proposed use of RICTs against these key safety principles is discussed below.

3.1.1 Key Principle 1: Evaluation of Compliance with Current Regulations

As stated in 10 CFR 50.36(c)(2), "[l]imiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

When the necessary redundancy is not maintained (e.g., one train of a two-train system is inoperable), the TSs permit a limited period of time to restore the inoperable train to OPERABLE status and/or take other remedial measures. If these actions are not completed within the CT, the TSs normally require that the plant exit the mode of applicability for the LCO. With one train of a two-train system inoperable, the TS safety function is accomplished by the remaining OPERABLE train. In the current TSs, the CT is specified as a fixed time period (termed the "front stop"). The addition of the option to determine the CT in accordance with the RICT Program would allow an evaluation to determine a configuration-specific CT. The evaluation would be done in accordance with the methodology prescribed in NEI 06-09, Revision 0-A. The RICT is limited to a maximum of 30 days (termed the "back stop") and can only be used when there is no TS loss of safety function. The CTs in the current TSs were established using experiential data, risk insights, and engineering judgement.

When the necessary redundancy is not maintained and the system loses the capability to perform its safety function(s) without any further failures (e.g., two trains of a two-train system are inoperable), there is a TS loss-of-function and the plant must exit the mode of applicability for the LCO, or take remedial actions, as specified in the TSs. A configuration-specific RICT may not be determined and used following a TS loss-of-function because the system has lost the capability to perform its safety function(s). With the incorporation of the RICT Program, the required performance levels of equipment specified in LCOs are not changed. Only the required CTs for the required actions are modified by the RICT Program.

The new TSs providing requirements for the RICT Program will be contained in the Administrative Controls Section of the TSs. The program specifies that the RICT Program be based on NEI 06-09, Revision 0-A, and specifies additional requirements to ensure the key principles described in this safety evaluation are satisfied. The NRC's regulations at 10 CFR 50.36(c)(5) do not provide prescriptive requirements for the content of Administrative controls programs.

In summary, the NRC staff concludes that the new TS describing the RICT Program provides the necessary controls relative to the RICT Program to assure operation of the facility in a safe manner, and thus satisfies 10 CFR 50.36(c)(5).

The minimum required performance levels of SSCs are not changed by the incorporation of the option to calculate a RICT. The remedial actions applicable when an LCO is not met are not changed by the incorporation of the RICT Program. Therefore, the NRC staff concludes that the TS, as modified, continue to satisfy the requirements in 10 CFR 50.36(c)(2).

3.1.1.1 Key Principle 1 Conclusions

Sections 3.1.2 and 3.1.3 of this safety evaluation provides an evaluation of the defense-in-depth and safety margin considerations associated with the RICT Program. For the reasons described in that section and for the reasons discussed above, the NRC staff concludes that the requirements of 10 CFR 50.36 are satisfied. This ensures that the plant will be operated in accordance with the design and is safe.

As such, the NRC staff finds that the proposed changes meet the first key safety principle of RG 1.174, Revision 3, and RG 1.177, Revision 1.

3.1.2 *Key Principle 2: Evaluation of Defense-in-Depth*

Defense-in-depth is an approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense-in-depth includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures.

As discussed throughout RG 1.174, consistency with the defense-in-depth philosophy is maintained by the following:

- 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.

- Maintain multiple fission product barriers.

- Preserve sufficient defense against human errors.

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

The proposed change represents a technical approach to balance the redundant and diverse key safety functions that provide avoidance of core damage, avoidance of containment failure, and consequence mitigation. The three-tiered approach to risk-informed TS CT changes described in RG 1.177 provides additional assurance that defense-in-depth will not be significantly impacted by such changes to the licensing basis. The licensee is proposing no changes to the design of the plant or any operating parameter, no new operating configurations, and no new changes to the design-basis in the proposed changes to the TS.

The effect of the proposed changes when implemented will be that the RICT Program 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 are inoperable). A configuration-specific RICT may not be determined and used following a TS loss-of-function when the system has lost the capability to perform its safety function(s). These restrictions on TS loss-of-function or inoperability of all required trains of a system ensure that consistency with the defense-in-depth 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. In some cases, the RICT Program may use compensatory actions to reduce calculated risk in some configurations. Where credited in the PRA, these actions are incorporated into station procedures or work instructions and have been modeled using appropriate human reliability considerations. Application of the RICT Program determines the risk significance of plant configurations. It also permits the operator to identify the equipment that has the greatest effect on the existing configuration risk. With this information, the operator can manage the out-of-service duration and determine the consequences of removing additional equipment from service.

The application of the RICT Program places high value on key safety functions and works to ensure they remain a top priority over all plant conditions. The RICT will be applied to extend CTs on key electrical power distribution systems. Failures in electrical power distribution systems can simultaneously affect multiple safety functions; therefore, potential degradation to defense-in-depth during the extended CTs are discussed further below.

3.1.2.1 *Use of Compensatory Measures to Retain Defense-in-Depth*

Application of the RICT Program provides a structure to assist the operator in identifying effective compensatory actions for various plant maintenance configurations to maintain and manage acceptable risk levels. NEI 06-09, Revision 0-A, addresses potential compensatory actions and RMA measures by stating, in generic terms, that compensatory measures may include but are not limited to the following:

- Reduce the duration of risk-sensitive activities.
- Remove risk-sensitive activities from the planned work scope.
- Reschedule work activities to avoid high risk-sensitive equipment outages or maintenance states that result in high-risk plant configurations.
- Accelerate the restoration of out-of-service equipment.
- Determine and establish the safest plant configuration.

NEI 06-09, Revision 0-A, requires that compensatory measures be initiated when the PRA calculated RMA time is exceeded, or for preplanned maintenance for which the RMA time is expected to be exceeded, RMAs shall be implemented at the earliest appropriate time. Therefore, quantitative risk analysis, the qualitative considerations, and the prohibition on loss of all trains of a required system assure a reasonable balance of defense-in-depth is maintained to ensure protection of public health and safety. The NRC staff finds that this proposed change meets the second key safety principle of RG 1.177 and is, therefore, acceptable.

3.1.2.2 Evaluation of Electrical Power Systems

According to the Turkey Point 3 and 4 UFSAR, the licensee indicates that the Units are designed such that the safety functions are maintained assuming a single failure within the electrical power system. By incorporating an electrical power supply perspective, this concept is further reflected in a number of principal design criteria for Turkey Point 3 and 4. Single failure requirements are typically suspended for the time that a plant is not meeting an LCO (i.e., in an ACTION statement). This section considers the plant configurations from a defense-in-depth perspective.

As discussed in Section 8.1 of the UFSAR, the following normal, standby and emergency power sources are available for Turkey Point 3 and 4:

- The source of auxiliary power during normal operation is the main generator and switchyard. The auxiliary transformer is connected to the generator isolated phase bus and the C bus transformer is connected to the switchyard. Both supply power to the 4.16 kilovolt (kV) system.

- Standby power during unit startup, shutdown and after unit trip is supplied from a startup transformer and a C bus transformer, which are connected to the switchyard 240 kV bus and feed the 4.16 kV system.

- Four EDGs supply emergency power. Each EDG is connected to a separate power train, two per unit. With any credible single failure, the EDGs are capable of assuring a safe shut down of both units with a loss of offsite power concurrent with Maximum Hypothetical Accident conditions in one unit.

- Emergency power for vital instrumentation and controls is supplied from four 125 volt (V) direct current (DC) station batteries. Each is capable of feeding its associated load for two hours without charging. A spare 125 V DC Station Battery is also provided which can be substituted for any of the four 125 V DC Station Batteries to allow for maintenance or testing.

- For each unit, a non-safety related 125 V DC bus provides power to the non-safety related C-bus 4.16 kV and 480 V switchgear, C-bus transformer relay panels and the turbine emergency oil pumps.

The licensee has requested to use the RICT Program to extend the existing CT for the following TS 3.8, "Electrical Power Systems," conditions. The NRC staff's evaluation of the proposed changes considered a number of potential plant conditions allowed by the proposed RICTs. The NRC staff also considered the available redundant or diverse means to respond to various plant conditions. In these evaluations, the NRC staff examined the safety significance of different plant conditions resulting in both shorter and longer CTs. The plant conditions evaluated are discussed in more detail below.

3.1.2.2.1 TS 3.8.1.1, AC Sources – Operating

TS 3.8.1.1 identifies the AC electrical power sources required to be operable during plant operations. The licensee proposed to add a RICT to TS 3.8.1.1, ACTIONs a, b, and f.

<u>TS 3.8.1.1 ACTION a</u>

The licensee proposed to apply RICTs to ACTION a, as shown in bold below:

> With one of two startup transformers or an associated circuit inoperable, demonstrate the OPERABILITY of the other startup transformer and its associated circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the inoperable startup transformer is the associated startup transformer and became inoperable while the unit is in MODE 1, reduce THERMAL POWER to ≤30% RATED THERMAL POWER within 24 hours, or restore the inoperable startup transformer and associated circuits to OPERABLE status within the next 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. If THERMAL POWER is reduced to ≤30% RATED THERMAL POWER within 24 hours or if the inoperable startup transformer is associated with the opposite unit restore the startup transformer and its associated circuits to OPERABLE status within 30 days of the loss of OPERABILITY, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. If the inoperable startup transformer is the associated startup transformer and became inoperable while the unit was in MODE 2, 3, or 4 restore the startup transformer and its associated 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 6 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

According to LCO 3.8.1.1, two startup transformers and their associated circuits (equivalent to the two offsite circuits required by GDC 17) are required to be operable. While in Mode 1, if the inoperable startup transformer is the associated startup transformer, thermal power is required to be reduced to ≤30% rated thermal within 24 hours, or the inoperable startup transformer and associated circuits are required to be restored to operable status within the next 48 hours. Therefore, according to this LCO, if a Unit is in Mode 1, the associated startup transformer and its associated circuits are required to be restored within 72 hours (24 hours+48 hours) to avoid reduction in thermal power. The addition of an option to calculate a RICT would allow for a longer CT of up to 30 days. Additionally, if the Unit is in Mode 2, 3, or 4, and the associated startup transformer is inoperable, the inoperable startup transformer and associated circuits are required to be restored to operable status within 24 hours. Similar to the Mode 1, the addition of the RICT option would allow for a longer CT of up to 30 days.

In the letter dated October 30, 2017 (in response to EEOB RAI-2), the licensee stated that a minimum of one startup transformer is required in order to meet the design success criteria. The licensee defined the design success criteria as, "[p]roviding offsite power to the required safety related loads to achieve and maintain safe shutdown (hot shutdown – STS Mode 4) of the units and provide power to the required safety related loads during a design basis accident on one unit." Because, the plant has two startup transformers, the required design success criteria can still be met with the remaining startup transformer. According to GL 80-30, it is not necessary to consider an additional single-failure when a plant is operating within a TS action statement.

In addition, in the letter dated October 30, 2017 (in response to EEOB RAI-1), the licensee stated that it will implement an RMA when in TS 3.8.1.1, ACTION a. The RMA will ensure that feedwater trains are fully operable, all EDGs are operable and protected, and verify that the onsite AC power distribution feed from the opposite unit startup transformer is operable.

Considering that the design success criteria will be met and that an RMA will be implemented during the RICT, the NRC staff finds that the plant will continue to have adequate defense-in-depth while in TS 3.8.1.1, ACTION a. Therefore, the proposed changes to TS 3.8.1.1, ACTION a are acceptable.

<u>TS 3.8.1.1, ACTION b</u>

The licensee proposed to apply a RICT to ACTION b, as shown in bold below:

> With one of the required diesel generators inoperable, demonstrate the OPERABILITY of the above required startup transformers and their associated circuits by performing Surveillance Requirement 4.8.1.1.1.a 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 preventative maintenance or testing, demonstrate the OPERABILITY of the remaining required diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours, unless the absence of any potential common mode failure for the remaining diesel generators is determined. If testing of remaining required diesel generators is required, this testing must be performed regardless of when the inoperable diesel generator is restored to OPERABILITY. Restore the inoperable diesel generator to OPERABLE status within 14 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 COLD SHUTDOWN within the following 30 hours.

**72 hours if inoperability is associated with Action Statement 3.8.1.1.c.

According to LCO 3.8.1.1, three EDGs are required to be operable—for Unit 3: EDG 3A and EDG 3B, and EDG 4A or EDG 4B; for Unit 4: EDG 4A and EDG 4B, and EDG 3A or EDG 3B. According to this ACTION b, an inoperable EDG must be restored within 14 days (footnote ** requires restoration of the inoperable EDG within 72 hours when the inoperability is associated with ACTION c. The NRC staff's evaluation of the impact to ACTION c from the proposed change to ACTION b is below). The addition of an option to calculate a RICT would allow for a longer CT of up to 30 days.

In the letter dated October 30, 2017 (in response to EEOB RAI-2), the licensee stated that a minimum of one EDG per unit is required to meet the design success criteria. The licensee defined the design success criteria as, "[p]roviding onsite power to the required safety related loads during a Loss of Offsite Power (LOP) to achieve and maintain safe shutdown (hot shutdown – STS Mode 4) of the units and provide power to the required safety related loads during a design basis accident on one unit without availability of offsite power." Because the plant has two EDGs per unit, the required design success criteria can be still met with the remaining EDGs. According to GL 80-30, it is not necessary to consider an additional single-failure when a plant is operating within a TS action statement.

In addition, in the letter dated October 30, 2017 (in response to EEOB RAI-1), the licensee stated that it will implement an RMA when in TS 3.8.1.1, ACTION b. The RMA will ensure that the other EDGs are operable and protected, and that the unaffected AC-trains are protected.

Considering that the design success criteria will be met during the RICT, and that an RMA will be implemented during the RICT, the NRC staff finds that the plant will continue to have adequate defense-in-depth while in TS 3.8.1.1, ACTION b. Therefore, the proposed change to TS 3.8.1.1, ACTION b is acceptable.

<u>TS 3.8.1.1 ACTION c:</u>

TS 3.8.1.1, ACTION c states:

> With one startup transformer and one of the required diesel generators inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a on the remaining startup transformer and associated circuits within one 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 preventive maintenance or testing, demonstrate the OPERABILITY of the remaining required diesel generators by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable diesel generator does not exist on the remaining required diesel generators, unless the diesel generators are already operating; restore one of the inoperable sources to OPERABLE status in accordance with Action Statements a and b, as appropriate. If testing of remaining required diesel generators is required, this testing must be performed regardless of when the inoperable diesel generator is restored to OPERABILITY. Notify the NRC within

> 4 hours of declaring both a start-up transformer and diesel generator inoperable. Restore the other A.C. power source (startup transformer or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 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.

The licensee proposed no changes to ACTION c. However, should the licensee enter ACTION c, the licensee would also enter either or both of ACTION a and ACTION b, as applicable at any given time. In the letter dated October 30, 2017 (in response to EEOB RAI-2), the licensee stated that a minimum of one startup transformer and one EDG (per unit) is required to meet the design success criteria, which is the same as that for ACTION b. The RMA for ACTION c will be similar that of ACTION a or ACTION b. The NRC staff finds that at a minimum of one startup transformer and 1 EDG (per unit) will remain available, assuming no additional single failure.

The NRC staff finds that although the licensee is not proposing any changes to ACTION c, the actions similar to either or both of ACTION a or ACTION b would be applicable at any given time. Because of this, the licensee will implement RMAs similar to those in ACTION a or ACTION b. Therefore, the NRC staff finds that the implementation of RMAs for ACTION c are acceptable based on the similar justifications provided for the cases of ACTION a or ACTION b described above.

<u>TS 3.8.1.1, ACTION f</u>

The licensee proposed to apply a RICT to ACTION f, as shown in bold below:

> With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two startup transformers and their associated circuits by performing the requirements of Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least 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. Restore all required diesel generators to OPERABLE status within 14 days from time of initial loss **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.

In the event of two of the three required EDGs are inoperable, TS 3.8.1, ACTION f directs the licensee to restore at least one of the inoperable EDGs within 2 hours or to shut down the plant. If one of the inoperable EDGs is restored within 2 hours, the licensee has up to 14 days to restore all of the required EDGs to operable status prior to being required to shut down the plant. The licensee proposes to allow for the option to calculate a RICT as an alternative to the 14-day CT.

Because the RICT would only be allowed after one of the two required EDGs is restored (i.e., only one of the required EDGs is inoperable), the proposed change is the same as that requested for ACTION b. For the same reasons discussed above for ACTION b, the NRC staff finds the proposed change to TS 3.8.1.1, ACTION f acceptable.

3.1.2.2.2 TS 3.8.2, DC Sources – Operating

TS 3.8.2.1 identifies the DC electrical power sources required to be operable during plant operations. The licensee proposed to add an option to calculate a RICT to TS 3.8.2.1, ACTIONs a and b.

TS 3.8.2.1, ACTION a

The licensee proposed to apply an optional RICT to ACTION a, as shown in bold below:

> With one or more of the required battery chargers OPERABLE but not capable of being powered from its associated OPERABLE diesel generator(s), restore the capability within 72 hours **or in accordance with the Risk Informed Completion Time Program,** or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

According to LCO 3.8.2.1, the following DC electrical sources are required, in part, to be operable:

> 125-volt D.C. Battery Bank 3A or spare battery bank D-52 and associated full capacity charger(s)
>
> 1) 3A1 powered by MCC 3C with EDG 3A OPERABLE, or
> 2) 3A2 powered by MCC 4D with EDG 4A and 4B OPERABLE, or
> 3) 3A1 powered by MCC 3C with EDG 3A OPERABLE and 3A2 powered by MCC 4D with EDG 4A and 4B OPERABLE,

Similar requirements apply to battery banks 3B, 4A, and 4B.

According to the above, one or both battery chargers connected to each DC bus and fed from their respective MCC (which in turn is fed from its respective EDG) are required to be operable.

In the letter dated October 30, 2017 (in response to EEOB RAI-2), the licensee stated that a minimum of one battery charger per operable DC bus is required in order to meet the design success criteria. The licensee defined the design success criteria as, "[p]roviding sufficient DC power for safe shutdown and mitigation and control of accident conditions." Based on the plant electrical distribution system, an inoperable EDG can impact operability of only one-of-two required chargers on any of the required DC buses. Since the plant has two full-capacity battery chargers per operable DC bus, the required design success criteria can be met with the remaining battery charger. According to GL 80-30, it is not necessary to consider an additional single-failure when a plant is operating within a TS action statement.

In addition, in the letter dated October 30, 2017 (in response to EEOB RAI-1), the licensee stated that it will implement an RMA when in TS 3.8.2.1, ACTION a. The RMA will limit the immediate discharge of the affected battery, ensure the battery float voltage is equal to or greater than the minimum required float voltage, and ensure that the other battery chargers and associated electrical distribution system are operable and protected.

Considering that the design success criteria will be met and that an RMA will be implemented during the RICT, the NRC staff finds that the plant will continue to have adequate defense-in-depth while in TS 3.8.2.1, ACTION a. Therefore, the proposed change to TS 3.8.2.1, ACTION a is acceptable.

TS 3.8.2.1 ACTION b

The licensee proposed to apply a RICT to ACTION b, as shown in bold below:

> With one of the required battery banks inoperable, or with none of the full-capacity chargers associated with a battery bank OPERABLE, restore all battery banks to OPERABLE status and at least one charger associated with each battery bank to OPERABLE status within two hours* **or in accordance with the Risk Informed Completion Time Program,** or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.
>
> *Can be extended to 24 hours if the opp[o]site unit is in MODE 5 or 6 and each of the remaining required battery chargers is capable of being powered from its associated diesel generator(s).

According to the above, either the required battery bank or one of the two associated battery chargers should be operable.

In the letter dated October 30, 2017 (in response to EEOB RAI-2), the licensee stated that a minimum of one battery and one charger per operable DC bus is required in order to meet the design success criteria. The licensee defined the design success criteria as, "[p]roviding sufficient DC power for safe shutdown and mitigation and control of accident conditions." The plant has a spare battery and a spare charger (with a capacity larger than any required battery bank and its associated charger) that can be connected to the required DC bus within 2 hours. According to UFSAR Chapter 8, each battery is capable of feeding its associated load for 2 hours without charging. The plant can meet the required design success criteria for the first 2 hours from either of two original DC sources (a battery bank or the associated full-capacity battery charger with only one source considered inoperable), and after 2 hours, from the spare battery bank or the spare battery charger. According to GL 80-30, it is not necessary to consider an additional single-failure when a plant is operating within a TS action statement.

Considering that the design success criteria will be met, and that an RMA will be implemented during the RICT, the NRC staff finds that the plant will continue to have adequate defense-in-depth while in TS 3.8.2.1, ACTION b. Therefore, the proposed change to TS 3.8.2.1, ACTION b is acceptable.

3.1.2.2.3 TS 3.8.3, Onsite Power Distribution System

TS 3.8.3.1 identifies the operability requirements for the onsite power distribution system during plant operations. The safety-related onsite AC power distribution system is as follows:

4.16 kV System

For each unit there are three safety-related 4.16 kV switchgear. Two of the switchgear, labeled as "A" and "B," provide power to the A and B trains of Engineered Safety Features, respectively, in each unit. The third safety-related 4.16 kV switchgear, labeled as "D" switchgear, is utilized as a swing bus. It can be manually aligned to either the A or B 4.16 kV bus of its respective unit. When the 4.16 kV swing switchgear is connected to either A or B 4.16 kV bus, it is considered an extension of that power supply bus (part of that train).

480 V System

For each unit there are five safety-related 480 V load center buses, four of which are arranged in double-ended load center configuration. Each of the four double-ended load center buses is fed from its associated unit by a separate load center transformer rated at 1000 kV-amp, 4160-480V. The two transformers of each double-ended unit are energized from different 4.16 kV buses (Load centers A and C are fed from Train A and Load Centers B and D are fed from Train B). This arrangement ensures the availability of equipment associated with a particular function in the event of loss of one 4.16 kV bus. The fifth safety-related 480 V load center in each unit is a swing load center, which can swing between load center C and D of its associated unit. As such, they do not have dedicated transformers. These load centers are labeled as 3H for Unit 3 and 4H for Unit 4. When the 480 V swing load center is connected to either 480 V supply bus, it is considered to be an extension of that supply bus. The control logic contains interlocks to prevent parallel connection to load centers 3C (4C) and 3D (4D). Safety-related 480 V MCCs are fed from the 480 V safety-related load center buses. Train A includes 480 V MCC buses A (Unit 4 only), C and D. Train B includes 480 V MCC buses B and D.

The licensee proposed to add an option to calculate RICTs to TS 3.8.3.1, ACTION a, Tables 3.8-1 and 3.8-2 associated with ACTION b, ACTION c, and ACTION d.

3.1.2.2.3.1 TS 3.8.3.1, ACTION a and ACTION b, Tables 3.8-1 and 3.8-2

According to LCO 3.8.3.1, as it relates to ACTIONs a and b, the following electrical busses* shall be energized in the specified manner with the tie breakers open between redundant busses within the Unit** and between the busses of Turkey Point 3 and 4:

a. One train of A.C. Busses consisting of:
    1) 4160-Volt Bus A,
    2) 480-Volt Load Center Busses A, C and H***, and
    3) 480-Volt Motor Control Center Busses A (Unit 4 only), C and D***,
b. One train of A.C. Busses consisting of:
    1) 4160-Volt Bus B
    2) 480-Volt Load Center Busses B, D and H***, and
    3) 480-Volt Motor Control Center Busses B and D***
c. One opposite unit train of AC busses consisting of either:
    1) 4160-Volt Bus A, 480-Volt Load Center Busses A, C and H***, and 480-Volt Motor Control Center Busses A (Unit 4 only), C and D***, or
    2) 4160-Volt Bus B, 480-Volt Load Center Busses B, D and H***, and 480-Volt Motor Control Center Busses B and D***.

*For Motor Control Center busses, vital sections only.

**With the opposite unit in MODE 5 or 6, its 480-Volt Load Center can be cross-tied under conditions specified in Specification 3.8.3.2.a.

***Electrical bus can be energized from either train of its unit and swing function to opposite train must be OPERABLE for the Unit(s) in MODES 1, 2, 3, and 4.

TS 3.8.3.1 ACTION a

The licensee proposed to apply a RICT to ACTION b, as shown in bold below:

> With one of the required trains (3.8.3.1 a., b., and c) of A.C. emergency busses not fully energized (except for the required [load centers] LC's and MCC's associated with the opposite unit), reenergize the train 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.

In the letter dated October 30, 2017 (in response to EEOB RAI-2), the licensee identified the following minimum equipment required to be operable in order to meet the design criteria:

One train per Unit consisting of:

- 4160-Volt Bus A,
- 480V Load Centers A, C, and H,
- 480V MCC A (Unit 4 only) and C

Or

- 4160-Volt Bus B,
- 480V Load Centers B, D, and H,
- 480V MCC B and D

The licensee defined the design success criteria as, "[p]roviding sufficient power for safe shutdown and mitigation and control of accident conditions."

According to above, in order to meet the design success criteria, only one train of AC busses are required to be operable (i.e., a redundant train can provide adequate power to meet the design success criteria). According to GL 80-30, it is not necessary to consider additional single-failure when a plant is operating within a TS action statement.

In addition, in the letter dated October 30, 2017 (in response to EEOB RAI-1), the licensee stated that it will implement an RMA when in TS 3.8.3.1, ACTION a. The RMA will consist of protecting the unaffected AC trains.

Considering that the design success criteria will be met, and that an RMA will be implemented during the RICT, the NRC staff finds that the plant will continue to have adequate defense-in-depth while in TS 3.8.3.1, ACTION a. Therefore, the proposed change to TS 3.8.3.1, ACTION a is acceptable.

TS 3.8.3.1 ACTION b, Tables 3.8-1 and 3.8-2

TS 3.8.3.1, ACTION b states:

> With any of the required [load centers] LC's and/or MCC's associated with the opposite unit inoperable, restore the inoperable LC or MCC to OPERABLE status

in accordance with Table 3.8-1 or Table 3.8-2 as applicable or place the unit in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

The licensee proposed to apply an option to calculate RICTs for the load centers and MCCs listed in Tables 3.8-1 and 3.8-2.

In the letter dated October 30, 2017 (in response to EEOB RAI-2), the licensee identified the following minimum equipment required to be operable in order to meet the design criteria:

One train of load centers and MCCs:

- 480V Load Centers A, C, and H, and
- 480V MCC A (Unit 4 only) and C

Or

- 480V Load Centers B, D, and H, and
- 480V MCC B and D

The licensee defined the design success criteria as, "[p]roviding sufficient power for safe shutdown and mitigation and control of accident conditions."

According to above, in order to meet design success criteria, only one train of load centers and MCCs are required to be operable (i.e., redundant train can provide adequate power to meet the design success criteria). According to GL 80-30, it is not necessary to consider an additional single-failure when a plant is operating within a TS action statement.

In addition, in the letter dated October 30, 2017 (in response to EEOB RAI-1), the licensee stated that it will implement an RMA when in TS 3.8.3.1, ACTION b. The RMA will consist of, "[n]o planned maintenance on required systems, subsystems, trains, components, and devices that depend on the remaining LCs/MCCs."

Considering that the design success criteria will be met, and that an RMA will be implemented during the RICT, the NRC staff finds that the plant will continue to have adequate defense-in-depth while in TS 3.8.3.1, ACTION b. Therefore, the proposed change to TS 3.8.3.1, ACTION b is acceptable.

3.1.2.2.3.2 TS 3.8.3.1, ACTION c

According to LCO 3.8.3.1, as it relates to ACTION c, the following 120 V vital panels shall be energized:

d. 120 Volt AC Vital Panel 3P06 and 3P21 energized from its associated inverter connected to D.C. Bus 3B.****

e. 120 Volt AC Vital Panel 4P06 and 4P21 energized from its associated inverter connected to D.C. Bus 3B.****

f. 120 Volt AC Vital Panel 3P07 and 3P22 energized from its associated inverter connected to D.C. Bus 3A.****

g. 120 Volt AC Vital Panel 4P07 and 4P22 energized from its associated inverter connected to D.C. Bus 3A.****

h. 120 Volt AC Vital Panel 3P08 and 3P23 energized from its associated inverter connected to D.C. Bus 4B.****

i. 120 Volt AC Vital Panel 4P08 and 4P23 energized from its associated inverter connected to D.C. Bus 4B.****

j. 120 Volt AC Vital Panel 3P09 and 3P24 energized from its associated inverter connected to D.C. Bus 4A.****

k. 120 Volt AC Vital Panel 4P09 and 4P24 energized from its associated inverter connected to D.C. Bus 4A.****

****A back-up inverter may be used to replace the normal inverter, provided the normal inverter on the same DC bus for the opposite unit is not replaced at the same time.

The licensee proposed to apply an option to calculate a RICT to ACTION c, as shown in bold below:

> With one A.C. vital panel either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) Reenergize the A.C. vital panel within 2 hours or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital panel from an inverter connected to its associated D.C. bus within 24 hours **or in accordance with the Risk Informed Completion Time Program,** or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

In the letter dated October 30, 2017 (in response to EEOB RAI-2), the licensee identified the following minimum equipment required to be operable in order to meet the design criteria:

> Three Vital AC Panels each energized by its own inverter per unit. Each inverter can only supply power to one vital AC panel.

The licensee defined the design success criteria as, "[p]roviding sufficient power for safe shutdown and mitigation and control of accident conditions."

According to above, in order to meet design success criteria, only three-out-of-four vital AC panels (each energized by its own inverter) per unit is required to be energized. According to GL 80-30, it is not necessary to consider an additional single-failure when a plant is operating within a TS action statement.

In addition, in the letter dated October 30, 2017 (in response to EEOB RAI-1), the licensee stated that it will implement an RMA when in TS 3.8.3.1, ACTION c. The RMA will consist of, "[n]o planned maintenance on required systems, subsystems, trains, components, and devices that depend on the remaining vital AC panels and DC busses."

Considering that the design success criteria will be met, and that an RMA will be implemented during the RICT, the staff finds that the plant will continue to have adequate defense-in-depth while in TS 3.8.3.1, RA "c." Therefore, the proposed change to TS 3.8.3.1, RA "c" is acceptable.

3.1.2.2.3.3 TS 3.8.3.1, ACTION d

According to LCO 3.8.3.1, as it relates to ACTION d, the following 125 V DC buses shall be energized:

> l. 125 Volt D.C. Bus 3D01 energized from an associated battery charger and from Battery Bank 3A or spare battery bank D-52,
>
> m. 125 Volt D.C. Bus 3D23 energized from an associated battery charger and from Battery Bank 3B or spare battery bank D-52,
>
> n. 125 Volt D.C. Bus 4D01 energized from an associated battery charger and from Battery Bank 4B or spare battery bank D-52, and
>
> o. 125 Volt D.C. Bus 4D23 energized from an associated battery charger and from Battery Bank 4B or spare battery bank D-52

The licensee proposed to apply an option to calculate a RICT to ACTION d, as shown in bold below:

> With one D.C. bus not energized from its associated battery bank or associated charger, reenergize the D.C. bus from its associated 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 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.
>
> * Can be extended to 24 hours if the opposite unit is in MODE 5 or 6 and each of the remaining required battery chargers is capable of being powered from its associated diesel generator(s).

In the letter dated October 30, 2017 (in response to EEOB RAI-2), the licensee identified the following minimum equipment required to be operable in order to meet the design criteria:

> One DC Bus per unit powered by One Battery and One Battery Charger.

The licensee defined the design success criteria as, "[p]roviding sufficient power for safe shutdown and mitigation and control of accident conditions."

According to above, in order to meet design success criteria, only one of the two DC busses per unit is required to be energized. According to GL 80-30, it is not necessary to consider an additional single-failure when a plant is operating within a TS action statement.

In addition, in the letter dated October 30, 2017 (in response to EEOB RAI-1), the licensee stated that it will implement an RMA when in TS 3.8.3.1, ACTION d. The RMA will consist of, "[n]o planned maintenance on required systems, subsystems, trains, components, and devices that depend on the remaining DC busses."

Considering that the design success criteria will be met, and that an RMA will be implemented during the RICT, the NRC staff finds that the plant will continue to have adequate defense-in-depth while in TS 3.8.3.1, ACTION d. Therefore, the proposed change to TS 3.8.3.1, ACTION d is acceptable.

3.1.2.2.4 Electrical Systems Conclusion

The NRC staff reviewed the licensee's proposed changes to the Turkey Point 3 and 4 electrical power system TSs 3.8.1, 3.8.2, and 3.8.3 that would add the option to calculate RICTs in accordance with the licensee's proposed RICT Program for certain required actions.

Based on the evaluation provided above (i.e., based on safety margins and defense-in-depth consideration), the NRC staff finds that the proposed changes meet the intent of the design criteria described in 1967 Proposed GDC 39 and GDC 17 concerning availability, capacity, and capability of the electrical power systems since the proposed changes do not make any design bases changes. Therefore, operation of the plant will continue to have adequate defense in depth, and will ensure safety margins are not affected adversely by the implementation of the RICT Program. As such, the NRC staff concludes the proposed changes are acceptable and consistent with RG 1.174 and RG 1.177.

The NRC staff reviewed the licensee's proposed TS changes and supporting documentation. Based on the evaluations above, the staff finds that while the redundancy is not maintained (e.g., one train of a two train system is inoperable), the CT extensions in accordance with the RICT Program are acceptable because (a) the capability of the systems to perform their safety functions (assuming no additional failures) is maintained, and (b) the licensee's demonstration of identifying and implementing compensatory measures or RMAs, in accordance with the RICT Program, are appropriate to monitor and control risk.

3.1.2.3 Evaluation of Instrumentation and Control (I&C) Systems

The licensee has requested to use the RICT Program to extend the existing CT for the following TS conditions. The NRC staff's evaluation of the proposed changes considered a number of potential plant conditions allowed by the new TSs and considered what redundant or diverse means were available to assist the licensee in responding to various plant conditions. The plant conditions evaluated are discussed in more detail below.

The NRC staff reviewed the Turkey Point 3 and 4 UFSAR and generic Westinghouse design information to independently confirm defense-in-depth for the identified I&C systems. The RG1.177 defense-in-depth criteria applicable to the identified Turkey Point 3 and 4 I&C systems are:

- System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system (e.g., there are no risk outliers).

- Defenses against potential common cause failures (CCF) are maintained and the potential for the introduction of new common cause failure mechanisms is assessed.

- The intent of the plant's design criteria is maintained.

The licensee states in Chapter 7 of the Turkey Point 3 and 4 UFSAR that the reactor protection system (RPS) and engineered safety features (ESF) instrumentation functions, in combination with the prescribed setpoints, are designed to detect the plant conditions that deviate from the predefined safety envelope, and to trip the reactor, or actuate the corresponding ESFs, respectively. TS Table 3.3-1 identifies 20 protective features (termed FUNCTIONAL UNITS in the TS) that can individually generate the signal to trip the reactor. Table 3.3-2 specifies 9 ESF safety functions ranging from safety injection (SI) to control room ventilation isolation, and under each safety function, presents protective features that produce signals to actuate that safety function.

Inoperability of one or more channels of a given RPS or ESF protective feature results in the need to take.a required action. While in an LCO required action, the redundancy will be temporarily reduced, and the system reliability will be reduced accordingly. The NRC staff examined the Turkey Point design information from its UFSAR and generic Westinghouse design information of diverse protective features that can compensate the temporary relaxation of redundancy for risk informed LCO conditions for the identified I&C systems to independently confirm that under any given design-bases accident, there are alternative I&C protective features available to detect and actuate appropriate safety functions to mitigate the accident.

The licensee described the I&C redundancy in Chapter 7 of the UFSAR and further evaluated the protective feature diversities in Chapter 14. The NRC staff examined the protective features for each safety function under each applicable individual plant condition to determine that at least one diverse protective feature exists for every proposed change.

For each instrument the NRC staff verified that in all the applicable operating modes, the protective feature will still perform its intended function by ensuring the ability to detect and mitigate the associated event or accident. Additionally, the NRC staff verified that identified alternatives will maintain appropriate diversity. The following sections summarize the NRC staff's findings and conclusions.

3.1.2.3.1 TS 3.3.1, Reactor Trip System Instrumentation

The LCO 3.3.1 requires that "[a]s a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE." The licensee proposed to apply an option to calculate RICTs to ACTIONs 1 and 10. ACTIONs 1 and 10 are applicable to Function Unit 1, "Manual Reactor Trip," and Function Unit 19, "Reactor Trip Breakers," respectively.

<u>Table 3.3-1, Functional Unit 1, ACTION 1</u>

The licensee proposed to revise ACTION 1 for Functional Unit 1 to state:

> With the number of OPERABLE channels one less than 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 HOT STANDBY within the next 6 hours.

According to TS Table 3.3-1, Functional Unit 1 consists of two channels with only one channel required to fulfill the manual trip function. The required Minimum Channels OPERABLE for Function Unit 1 is two. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE required, one channel remains OPERABLE. The NRC staff finds that this one channel can fulfill the manual trip function. However, the redundancy of this function is degraded from 1-out-of-2 to 1-out-of-1.

Chapter 14 of the Turkey Point 3 and 4 UFSAR states that the "Manual Reactor Trip" mitigates multiple accidents, including small loss-of-coolant accident (LOCA), steam generator tube rupture, and large LOCA. The NRC staff grouped these accidents below with respect to alternate protective features available.

The first group includes small LOCA and steam generator tube rupture. For these accidents, the "Pressurizer Pressure (low) trip" is the alternative protective feature to the "Manual Reactor Trip." The NRC staff concludes that the diversity for the "Manual Reactor Trip" and subsequently the consistency with the defense-in-depth principle are maintained.

The second group includes a large number of accidents including large LOCA. For each of these accidents, the automatic reactor trip features and/or SI features are available as diverse mitigations. The NRC staff concludes that the diversity and, thus, the defense-in-depth principle is adequately maintained for this change. Therefore, the proposed change is acceptable.

Table 3.3-1, Functional Unit 19, ACTION 10

The licensee proposed to revise ACTION 10 for Functional Unit 19 to state:

> With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable, except for the time required for performing maintenance to restore the breaker to OPERABLE status.

Based on UFSAR Figure 7.2-5, the reactor trip mechanism consists of two redundant trains (Train A and Train B). Each train has two internally diverse trip features (undervoltage and shunt) connected with one reactor trip breaker component. If a trip condition occurs, the trip signal(s) controls the undervoltage and shunt trip coils to open trip breaker contacts. The Rod Control System then drops the rods to shut down the reactor. With one trip feature inoperable, the Reactor Trip System redundancy is not compromised: it relies on another trip feature and maintains two trains operable to fulfill the trip function. The NRC staff finds that the internal diversity within the reactor trip breaker is temporarily reduced.

As discussed in Chapter 7 of the UFSAR, the reactor trip breaker is diversified by the Anticipated Transient Without Scram Mitigating System Actuation Circuitry feature to trip the turbine and actuate the auxiliary feedwater system. The NRC staff concludes that the reactor trip breaker diversity is adequately maintained, and the risk-informed CT of this action complies with the defense-in-depth philosophy. Therefore, the proposed change is acceptable.

3.1.2.3.2 TS 3.3.2, ESFAS Instrumentation

The LCO 3.3.2 requires, in part, that the ESFAS instrumentation channels and interlocks shown in Table 3.3-2 be OPERABLE.

The licensee proposed the new ACTIONs 26 and 27 to apply RICTs to Functional Units in Table 3.3-2. New ACTION 26 would apply to the following Functional Units:

- Functional Unit 1.c, "Containment Pressure – High"
- Functional Unit 1.d, "Pressurizer Pressure – Low"
- Functional Unit 1.e, "High Differential Pressure Between Steam Line Header and any Steam Line"
- Functional Unit 1.f, "Steam Line flow--High," coincident with, "Steam Generator Pressure--Low"
- Functional Unit 2.b, "Containment Pressure--High-High," coincident with, "Containment Pressure--High"
- Functional Unit 6.b, "Stm. Gen. Water Level--Low-Low"

New ACTION 27 would apply to the following Functional Unit 1.a, "Manual Initiation," and Functional Unit 3.a.1, "Phase 'A' Isolation, Manual Initiation."

3.1.2.3.2.1 ACTION 26

The licensee proposed to replace ACTION 15 with proposed ACTION 26 for the Functional Units identified above. ACTION 15 states:

> With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST or TRIP ACTUATING DEVICE OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 26 would state:

> With one channel inoperable, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST or TRIP ACTUATING DEVICE OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours or in accordance with the Risk Informed Completion Time Program.

The NRC staff examined the applicable Functional Units in Table 3.3-2 individually to validate the acceptance of the temporary reduction in defense-in-depth caused by the one channel inoperable LCO conditions during the RICT periods.

<u>Functional Unit 1.c</u>

In accordance with Table 3.3-2, the coincidence logic for the Functional Unit 1.c is two-out-of-three. With one channel inoperable, the Functional Unit 1.c is still able to initiate the SI signal. The NRC staff finds that the reliability and/or redundancy of this protective function is temporarily degraded from 2-out-of-3 to 2-out-of-2.

Chapter 14 of the Turkey Point 3 and 4 UFSAR shows that Functional Unit 1.c mitigates the accidents identified below. The NRC staff grouped these accidents with respect to alternate protective functions available.

Group 1: Group 1 accidents include the large LOCA. The NRC staff finds that the protective alternatives to mitigate this accident include the automatic SI initiated by the Pressurizer pressure (low) and manual SI initiation. The NRC staff concludes that the temporary degradation on reliability/redundancy due to extended RICT is acceptable for this action, because the two alternative protective features maintain adequate diversity and thus, defense in depth, under this accident condition.

Group 2: Group 2 accidents include the steamline break accident. The NRC staff finds the following alternative SI features that can be used to mitigate this accident:

1. Automatic SI:
   a. Pressurizer pressure (low)
   b. Steamline pressure (low in one loop)
   c. Differential pressure between steamlines (high)
   d. T_{avg} (low-low)
   e. Steamline flow (high) in two steamlines coincident with T_{avg} (low-low)
   f. Steamline flow (high) in two steamlines coincident with steamline pressure (low)
2. Manual initiation

The NRC staff concludes that the number of alternative SI features under this accident condition assure the abundant diversity and, subsequently, maintains the consistency with the defense-in-depth philosophy. Therefore, the NRC staff finds that the proposed change is acceptable.

Functional Unit 1.d

In accordance with TS Table 3.3-2, the coincidence logic for Functional Unit 1.d is two-out-of-three. The NRC staff notes that with one channel inoperable, the Functional Unit 1.d is still able to initiate the SI signal.

Chapter 14 of the UFSAR shows that this safety feature can actuate three safety functions: reactor trip, SI, and steamline isolation. The accidents these safety features mitigate and alternative safety features for each safety function are discussed below. The NRC staff grouped the accidents with respect to the safety function and alternate protective features available.

A. Reactor trip safety function

The first group of accidents includes small LOCA and steam generator tube rupture. The alternative protective feature to mitigate these accidents is "Manual Trip" (Table 3.3-1, Functional Unit 1). This is the only alternative to the "Pressurizer Pressure – Low" under these accident conditions. The NRC staff finds that the diversity is adequate because the proposed change does not change the diversity adequacy, as approved under present licensing basis. The NRC staff concludes that the defense-in-depth philosophy is reasonably maintained and that the change is acceptable under this group of accidents.

The second group of accidents include the steamline break. Multiple safety features can be used as alternatives to Function Unit 1.d, including the following:

1. Automatic Rx Trip:
   a. Power range neutron flux (high)
   b. Power range neutron flux rate (high positive rate)

   c. Overpower Delta T
   d. Overtemperature Delta T
   e. Containment pressure (high-1, via SI signal)
2. Manual trip

The NRC staff finds that the diversity is adequate and the defense-in-depth philosophy is reasonably maintained, and that the proposed change is acceptable for this group of accidents.

B. SI function

Functional Unit 1.d initiates the SI function to mitigate the following groups of accidents:

The first group includes the large LOCA. A number of protective features can be used as alternatives to the "Pressurizer Pressure – Low," which include the following:

1. Automatic SI:
   a. Containment pressure (high-1)
2. Manual Initiation

The NRC staff finds that the diversity is adequate, that the defense-in-depth philosophy is reasonably maintained, and that the proposed change is acceptable for this group of accidents.

The second group includes accidents of small LOCA and steam generator tube rupture. Chapter 7 of the UFSAR shows that the alternative to the "Pressurizer Pressure – Low" is "Manual Initiation." The NRC staff finds that the defense-in-depth philosophy is reasonably maintained as one diverse alternative to the "Pressurizer Pressure – Low" exists. The NRC staff finds that the change is acceptable under this group of accidents.

The third group includes the accident steamline break. Multiple protective features can be used as alternatives to the "Pressurizer Pressure – Low" under this accident condition, including the following:

1. Automatic SI:
   a. Containment pressure (high)
   b. Steamline pressure (low in one loop)
   c. Differential pressure between steamlines (high)
   d. Steamline flow (high) in two steamlines coincident with
   e. T_{avg} (low-low)
   f. Steamline flow (high) in two steamlines coincident with steamline pressure (low)
2. Manual Initiation

The NRC staff finds that the diversity is adequate, that the defense-in-depth philosophy is reasonably maintained, and that the proposed change is acceptable for this group of accidents.

The fourth group includes the accident feedwater line break. Multiple protective features can be used as alternatives to the "Pressurizer Pressure – Low" under this accident condition, including the following:

1. Automatic SI:
   a. Containment pressure (high)
   b. Steamline pressure (low in one loop)

   c. Differential pressure between steamlines (high)
   d. Steamline flow (high) in two steamlines coincident with T_{avg} (low-low)
   e. Steamline flow (high) in two steamlines coincident with steamline pressure (low)
2. Manual Initiation

The NRC staff finds that the diversity is adequate, that the defense-in-depth philosophy is reasonably maintained, and that the proposed change is acceptable for this group of accidents.

C. Steamline Isolation

Chapter 14 of the UFSAR shows that the accident mitigated by the "Pressurizer Pressure – Low" under the steamline isolation safety function is the steamline break." Chapter 14 identifies multiple protective features that can be used as alternatives to the "Pressurizer Pressure – Low," including the following:

1. Automatic Rx Trip:
   a. Steamline pressure (negative rate)
   b. Steamline flow (high) in two steamlines coincident with Tavg (low-low)
   c. Steamline flow (high) in two steamlines coincident with steamline pressure (low)

The NRC staff finds that the diversity and the defense-in-depth philosophy is reasonably maintained, and that the proposed change is acceptable under this group of accidents.

<u>Functional Unit 1.e</u>

In accordance with Table 3.3-2, the coincidence logic for the Functional Unit 1.e is two-out-of-three for each steam line. With one channel inoperable, Functional Unit 1.e is still able to initiate the SI signal. The reliability/redundancy is temporarily reduced during the extended RICT.

Chapter 14 of the UFSAR identifies accidents that this protective feature mitigates. The NRC staff grouped these accidents into two groups, then examined the alternative protective features for each group of accidents to determine the adequacy of the diversity.

The first group includes the accident steamline break. Multiple SI features can be used as alternatives to Functional Unit 1.e under this accident condition, including the following:

1. Automatic SI:
   a. Pressurizer pressure (low)
   b. Containment pressure (high)
   c. Steamline pressure (low in one loop)
   d. Steamline flow (high) in two steamlines coincident with T_{avg} (low-low)
   e. Steamline flow (high) in two steamlines coincident with steamline pressure (low)
2. Manual Initiation

The NRC staff finds that the diversity is adequate, that the defense-in-depth philosophy is reasonably maintained, and that the proposed change is acceptable under this group of accidents.

The second group includes accident of "Feedwater Line Break." A number of SI features can be used as alternatives to the Functional Unit 1.e under this accident condition, which include the following:

1. Automatic SI:
   a. Containment pressure (high)
   b. Steamline pressure (low in one loop)
2. Manual Initiation

The NRC staff finds that the diversity is adequate, that the defense-in-depth philosophy is reasonably maintained, and that the proposed change is acceptable for this group of accidents.

Functional Unit 1.f

Based on UFSAR Figure 7.2-5, the design of the Functional Unit 1.f shows two channels of steam flow detectors and one channel of pressure detector per steam generator (with three steam generators, in total). The one-out-of-two coincidence is used to generate the per-steam generator high flow signal, and two-out-of-three logic is used to initiate the high-flow signal for three steam generator lines. The two-out-of-three logic is used to initiate the low-pressure signal for the three steam generators. The two-out-of-two logic is then used to combine the high-flow signal and low-pressure signal to initiate the safety injection signal. The NRC staff notes that with one steam-flow channel inoperable, coincident with one steam-generator pressure channel inoperable, the safety injection logic described above is still able to initiate the SI signal. The reliability/redundancy is temporarily reduced during the extended RICT.

As discussed in UFSAR Chapter 14, this protective feature is used by both SI and steamline isolation functions to mitigate the steamline break accident. The accidents these safety features mitigate and alternative safety features for each safety function are discussed below. The NRC staff grouped the accidents with respect to the safety function and alternate protective features available.

For SI safety function, multiple alternatives can be used, including the following:

1. Automatic SI:
   a. Pressurizer pressure (low)
   b. Containment pressure (high)
   c. Steamline pressure (low in one loop)
   d. Steamline flow (high) in two steamlines coincident with T_{avg} (low-low)
   e. Differential pressure between steamlines (high)
2. Manual Initiation

The NRC staff finds that the diversity is adequate, that the defense-in-depth philosophy is reasonably maintained, and that the proposed change is acceptable under this group of accidents.

For the "Steamline Isolation" safety function, multiple alternatives can be used, including the following:

- Containment pressure (high-high)
- Steamline pressure (low)
- Steamline pressure (negative rate)

- Steamline flow (high) in two steamlines coincident with T_{avg} (low-low)"Pressurizer pressure (low)

The NRC staff finds that the diversity is adequate, that the defense-in-depth philosophy is reasonably maintained, and that the change is acceptable.

Functional Unit 2.b

According to TS Table 3.3-2, the coincidence logic for Functional Unit 2.b is two-out-of-three. The NRC staff notes that with one channel of "Containment Pressure – High-High" inoperable, and/or one channel of "Containment Pressure – High" logic inoperable, Functional Unit 2.b is still able to initiate Containment Spray signal. The reliability/redundancy is temporarily reduced during the extended RICT.

As discussed in UFSAR Chapter 14, this protective feature is used to initiate the containment spray safety function. This safety function mitigates all accident events that can potentially increase the containment pressure, such as LOCA and steamline break inside containment.

As discussed in UFSAR Chapter 7, a manual actuation is in place to actuate the containment spray in addition to the above automatic actuation. The NRC staff concludes that the diversity is adequate, that the defense-in-depth philosophy is reasonably maintained, and that this change is acceptable.

Functional Unit 6.b

According to TS Table 3.3-2, the coincidence logic for Functional Unit 6.b is two-out-of-three. The NRC staff notes that with one channel of "Steam Generator Water Level – Low-Low" inoperable, Functional Unit 6.b is able to initiate the auxiliary feedwater start signal. The reliability/redundancy is temporarily reduced during the extended RICT.

The licensee stated in Section 14 of the Turkey Point UFSAR that this protective feature initiates the Auxiliary Feedwater safety function. This safety function mitigates the accidents of "Positive Reactivity Insertion," "Loss of Reactor Coolant Flow," "Loss of Main Feedwater," "Loss of Condenser" and "Turbine trip." No alternative protective feature is identified to initiate the Auxiliary Feedwater except for the "Loss of Main Feedwater" accident which is diversified by the "Main Feedwater Pump Trip" safety feature.

The licensee stated that the Turkey Point plant has three steam generators. Each steam generator has its own Functional Unit 6.b "Steam Generator Water Level – Low-Low" signal. Each signal alone can start the Auxiliary Feedwater. The NRC staff finds that this design features an additional 1-out-of-3 redundancy on top of the 2-out-of-3 redundancy.

The NRC staff finds that the proposed change to this LCO condition and associated action do not change the diversity adequacy and the additional 1-out-of-3 redundancy. The NRC staff finds that the defense-in-depth philosophy is reasonably maintained and the change is acceptable.

3.1.2.3.2.2 ACTION 27

The licensee proposed to replace ACTION 17 with proposed ACTION 27 for Functional Unit 1.a and Function Unit 3.a.1. ACTION 17 states:

> With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 27 would state:

> With one channel inoperable, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program, or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

The NRC staff examined the applicable Functional Units in Table 3.3-2 individually to validate the acceptability of the temporary reduction in defense-in-depth caused by the one channel inoperable LCO conditions during the RICT periods.

Functional Unit 1.a

According to Table 3.3-2, the coincidence logic for Functional Unit 1.a is one-out-of-two. The NRC staff notes that with one channel inoperable, the Functional Unit 1.a is still able to initiate the SI signal. Its reliability and redundancy are temporarily reduced during the extended RICT.

Chapter 14 of the UFSAR identifies a set of accidents that Functional Unit 1.a mitigates. The NRC staff grouped these accidents with respect to alternative protective functions available.

The first group includes the accident large LOCA. The protective alternatives to mitigate this accident include automatic SI, initiated by the "Pressurizer pressure (low)" and "Containment pressure (high)" signals. The NRC staff finds the diversity of this safety feature is adequate for the proposed change.

The second group includes the accident small LOCA. The alternative protective feature to mitigate this accident is "Pressurizer pressure (low)." The NRC staff finds the diversity of this safety feature is adequate during the proposed RICT.

The third group includes the accident of steamline break. The alternative protective features to mitigate this accident are:

- Pressurizer pressure (low)
- Containment pressure (high)
- Steamline pressure (low in one loop)
- Differential pressure between steamlines (high)
- Steamline flow (high) in two steamlines coincident with T_{avg} (low-low)
- Steamline flow (high) in two steamlines coincident with steamline pressure (low)

The NRC staff finds the diversity of this safety feature is adequate for the proposed change.

The fourth group includes the accident of feedwater line break. The alternative protective features to mitigate this accident are:

- Containment pressure (high)
- Steamline pressure (low in one loop)
- Differential pressure between steamlines (high)

The NRC staff concludes the diversity of this safety feature is adequate for the proposed change.

The NRC staff finds that the proposed RICT for the Functional Unit 1.a ACTION 27 temporarily reduces the redundancy of this protective feature. Based on the above analysis, the NRC staff finds that adequate diversity of the protective features is maintained, and that the defense-in-depth philosophy is reasonably maintained for the proposed change to the Functional Unit 1.a, ACTION 27.

Functional Unit 3.a.1

According to Table 3.3-2, the coincidence logic for the Functional Unit 3.a.1 is one-out-of-two. The NRC staff notes that with one channel inoperable, the Functional Unit 3.a.1 is still able to initiate the containment isolation signal. Its reliability and redundancy are temporarily reduced during the proposed RICT.

Chapter 14 of the UFSAR shows that this "Manual Initiation" safety feature initializes the Phase "A" Containment Isolation safety function to mitigate all SI events. All SI safety features (six automatic SI features and one manual initiation feature – Functional Unit 1.a to 1.f) are used as its diversity. The NRC staff finds that the diversity of this safety feature is adequate and that the defense-in-depth is reasonably maintained for the proposed change to the Functional Unit 3.a.1, ACTION 27.

3.1.2.3.3 I&C Systems Conclusion

The Turkey Point 3 and 4 TS 3.3 "Instrumentation" LCOs were developed to assure that the plant maintains the necessary reliability, redundancy, and diversity, in compliance with the "reliability" criteria (1967 Proposed GDC 19); the "single failure" design criterion, as defined in Clause 4.2 of IEEE 279 (and 1967 Proposed GDC 20), and "Channel Bypass or Removal from Operation" criterion, as defined in Clause 4.11 of IEEE 279.

As discussed in Sections 3.1.2.3.1 and 3.1.2.3.2 above, the I&C safety functions identified as part of the licensee's proposed change maintain the capability to perform their safety functions when in a RICT. Therefore, the I&C system diversity configuration remains unchanged. Based on the evaluation presented above, the NRC staff concludes that the proposed changes to TS 3.3 comply with the diversity principle, as specified in defense-in-depth philosophy described in RG 1.174.

Under certain TS Conditions the single failure criterion cannot be met, because the inoperable channels can be functionally assumed to be removed from operation. The required action (e.g., placing the channel in trip) is the method chosen, based on the design, to comply with Clause 4.11. The RICT Program provides the necessary administrative controls to permit an extension of CTs and, thereby, delay reactor shutdown or other Required Actions. If risk is assessed and managed appropriately within specified limits and programmatic requirements, the NRC staff considers that the affected system operation reliability remains acceptable and is consistent with overall system reliability and risk considerations. Therefore, the NRC staff concludes that the proposed changes to TS 3.3 meet requirements defined in the plant's principal design criteria (i.e., 1967 Proposed GDC 19 and 20), as well as Clause 4.2 and Clause 4.11 of IEEE 279.

Because the licensee did not propose any changes to the design basis, the independence and the fail-safe principle remain unchanged. The licensee stated in the license amendment request (LAR) that the proposed changes did not include any TS loss-of-function conditions. However, it is recognized that while in an ACTION statement, redundancy of the given protective feature will be temporarily reduced, and, accordingly, the system reliability will be reduced. In the LAR, the licensee stated in the description of proposed changes to the instrumentation and control systems that at least one diverse means (e.g., other automatic features or manual action) to accomplish the safety functions (e.g., reactor trip, SI, or containment isolation) remain available during the use of the RICT. The NRC staff reviewed the licensee's proposed TS changes to assess the availability of the diverse means to accomplish the safety function(s). The NRC staff finds that the availability of diverse protective features provide sufficient defense-in-depth to accomplish the safety functions, allowing for the extension of CTs in accordance with the RICT Program.

The NRC staff reviewed the licensee's proposed TS changes and supporting documentation. The NRC staff finds that while the instrumentation and control redundancy is reduced, the CT extensions implemented in accordance with the RICT Program are acceptable because: (a) the capability of the instrumentation and control systems to perform their safety functions is maintained, (b) diverse means to accomplish the safety functions exist, and (c) the licensee will identify and implement RMAs to monitor and control risk in accordance with the RICT Program.

3.1.2.4 Key Principle 2 Conclusions

The LAR proposes to modify the TS requirements to permit extending selected CTs using the RICT Program in accordance with NEI 06-09, Revision 0-A. The NRC staff has reviewed the licensee's proposed TS changes and supporting documentation. 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 defense-in-depth provided that the licensee identifies and implements compensatory measures, as appropriate, during the extended CT in accordance with the RICT Program.

As discussed above in this safety evaluation, the NRC staff has further evaluated key safety functions in the proposed CT extensions and concluded that (1) the changes maintain the intent of the design criteria; (2) the specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences preserving system redundancy, independence, and diversity commensurate with the expected frequency, consequences of challenges to the system, and uncertainties; and (3) sufficient capacity and capability is maintained to assure that containment integrity and other vital functions are maintained in the event of postulated accidents, preserving the independence of barriers.

The NRC staff concludes that the proposed changes are consistent with the defense-in-depth philosophy with respect to the requirements in 1967 Proposed GDC 39 and GDC 17 concerning availability, capacity, and capability of the electrical power systems, and requirements in 1967 Proposed GDC 19 and 20, as well as Clause 4.2 and Clause 4.11 of IEEE 279 concerning availability, high reliability, testability, independence, fail safe, and function diversity of the instrumentation and control systems. The proposed changes are also consistent with 10 CFR 50.36(c)(2) because the lowest functional capability or performance levels of equipment required for safety is maintained. Therefore, the NRC staff concludes that the proposed changes are acceptable and consistent with the principle of defense-in-depth.

3.1.3 Key Principle 3: Evaluation of Safety Margins

Section 2.2.2 of RG 1.177, Revision 1, 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 (FSAR) are met or proposed revisions provide sufficient margin to account for analysis and data uncertainties.

The licensee is not proposing in this application to change any quality standard, material, or operating specification. In Attachment 2 of letter dated October 30, 2017, supplemented by letter dated February 15, 2018, the licensee proposed to add a new program, "Risk Informed Completion Time Program," in Section 6.0, "Administrative Controls (6.8.4)," of the TSs, which would require adherence NEI 06–09, Revision 0-A. The Risk Informed Completion Time Program states that use of a RICT is not permitted for entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE.

Acceptance criteria for operability of equipment are not changed and use of the RICT only when the system(s) retain(s) the capability to perform the applicable safety function(s) ensure that the current safety margins are retained. 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, Revision 0-A, is limited to the inoperable train, loop, or component. The reduced, but available, functionality may support a further increase in the CT consistent with the risk of the configuration. During this increased CT, the specified safety function is still being met by the operable train and, therefore, functionality to meet the design basis success criteria is maintained.

3.1.3.1 Key Principle 3 Conclusions

As discussed above, the NRC staff finds that the design-basis analyses for Turkey Point remain applicable. Although the licensee will be able to have design-basis equipment out-of-service longer than the current TS allow and the likelihood of successful fulfillment of the function will be decreased when redundant train(s) are not be available, the capability to fulfill the function will be retained. Any increase in unavailability because less equipment is available for a longer time is included in the RICT evaluation. Therefore, safety margins are not affected adversely by the implementation of the RICT Program. Based on the above, the NRC staff concludes that the proposed change meets the third key safety principle of RG 1.177 and is acceptable.

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

Proposed TS Section 6.8.4.p. "Risk Informed Completion Time Program," in Attachment 2 of letter dated October 30, 2017, states that the RICT "must be implemented in accordance with

NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b: Risk-Managed Technical Specifications (RMTS) Guidelines,' Revision 0-A, November 2006."

NEI 06-09, Revision 0-A, is 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 Chapter 16.1 of the SRP, RG 1.177, and RG 1.174. This approach addresses the calculated change in risk as measured by the ΔCDF and ΔLERF, as well as the ICCDP and ICLERP; the use of compensatory measures to reduce risk; and, the implementation of a CRMP to identify risk-significant plant configurations.

Below is the NRC staff's evaluation of the licensee's proposed changes against the three-tiered approach in RG 1.177.

3.1.4.1 Tier 1: PRA Capability and Insights

The first tier evaluates the impact of the proposed changes on plant operational risk. The Tier 1 review involves two aspects: (1) the technical 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.

3.1.4.1.1 PRA Quality

RG 1.174 states that the scope, level of detail, and technical adequacy of the PRA are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process. The NRC's safety evaluation as described in NEI 06-09, Revision 0-A, states that the PRA models should conform to the guidance in RG 1.200, Revision 1 (Reference 34). The current version is RG 1.200, Revision 2, which clarifies the current applicable ASME/ANS PRA standard is ASME/ANS RA-Sa-2009. The Turkey Point PRA model has used the current ASME/ANS PRA standard and RG 1.200 as noted below. The Turkey Point PRA model is composed of an Internal Events PRA (IEPRA) (including internal flooding) and a Fire PRA (FPRA). The licensee screened out all external hazard events, as described below, as insignificant contributors to RICT calculations. The Turkey Point PRA model with modifications is used as the CRMP model as described in Enclosure 8 of the LAR.

The Turkey Point PRA is maintained and updated under a PRA configuration control program in accordance with station procedures. Periodic reviews are conducted and updates are performed, as necessary, for plant changes including performance data, procedures, and modifications. In Enclosure 7, "PRA Model Update Process" of the LAR, the licensee stated that the reviews and updates will be performed by qualified personnel with independent reviews and approvals.

On February 24 and 25, 2016, the NRC staff and its contractors from the Pacific Northwest National Laboratory participated in a regulatory audit at the NextEra Energy Offices in Juno Beach, Florida, of the PRA quality information provided by the licensee by application dated December 23, 2014.

Internal Events PRA (including Internal Flooding)

The licensee's evaluation of the technical adequacy of its IEPRA model included a combination of peer reviews, self-assessments and use of an independent assessment (IA) team. A full scope peer review of the Turkey Point IEPRA was performed in July 2002 by the Combustion

Engineering Owners Group using NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance" (Reference 35), which pre-dated the ASME/ANS PRA standard and RG 1.200. Shortly after the PRA standard, ASME/ANS RA-Sa-2009 and RG 1.200, Revision 2 clarifications were issued, the licensee performed a self-assessment on the then-current model to identify gaps in the IEPRA against the current PRA standard. In April 2011, a focused-scope peer review on the human reliability analysis (HRA) element and the internal flooding portion of the PRA was performed by the Pressurized Water Reactor Owners Group using PRA standard, ASME/ANS-RA-Sa-2009, as clarified by RG 1.200, Revision 2. In October 2013, a focused-scope peer review of the common-cause-failure, Level 2, and interfacing system LOCA analyses was performed for IEPRA modeling that had been upgraded. By letter dated February 15, 2018, the licensee stated that an IA team had completed the Facts and Observations (F&Os) closure process, in accordance with the NRC-approved process in Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13, "Close Out of Facts and Observations" (Reference 36). The licensee did not provide the date of the F&O closure review, but Attachment 3 to the February 15, 2018, letter provided the results of the review. The NRC staff did not assess the implementation of the IA F&O Closure process because the F&Os were provided in their entirety.

Table 1 in Enclosure 2 to the LAR provides the full-scope F&Os, the gap assessment findings, and the focused-scope F&Os along with dispositions about how the F&O or gap assessment finding was resolved, or why the F&O or gap assessment finding was found not to have a significant impact for this application. The NRC staff reviewed each peer-review F&O and gap assessment finding along with the associated resolutions in the LAR and in the supplemental information supporting the LAR. The NRC staff requested additional information regarding the resolution to some of the F&Os, as discussed below.

F&Os AS-1, AS-2, AS-3, and AS-9, cite instances in which IEPRA success criteria appeared not to be properly modeled or developed. In APLA RAI 1, the NRC staff requested explanation for how the current IEPRA success criteria were determined and clarification that determination and documentation of success criteria was consistent with the PRA standard. In response to APLA RAI 1, the licensee explained that success criteria were primarily determined using Modular Accident Analysis program (MAAP) simulations with plant-specific information, but that in some cases plant approved references were used instead. The licensee explained that, after the 2002 full-scope peer review, the calculation and documentation of success criteria has been evaluated against ASME/ANS RA-Sa-2009 success criteria related Supporting Requirements, and the licensee has concluded that all associated Supporting Requirements are met at a Capability Category II level or higher. The licensee will estimate the RICT values based on PRA success criteria (e.g., minimum flow rates and time available for human actions) developed by the MAAP calculations, but the proposed Risk Informed Completion Time Program clarifies that at least one train will remain OPERABLE and, therefore, the capability to satisfy the design-basis success criteria shall continue to be met. The NRC staff concludes that the licensee has adequately resolved these F&Os because the licensee used an industry state-of-the-art tool (i.e., MAAP) and plant-specific information for determining PRA success criteria, and an extended CT will only be used if the design-basis success criteria will be available during that extended CT.

F&O 4-24, related to IEPRA Supporting Requirements FQ-C1, HR-G6, HRA-C1, and QU-C1, generated during the Fire Probabilistic Risk Assessment (FPRA) peer review, questioned whether the post initiator human error probabilities (HEPs) were reasonable given the scenario context of the required human action. Based on the licensee's disposition of F&O 4-24, it appeared to NRC staff that minimum joint HEPs had not been applied in the IEPRA. Therefore in APLA RAI 6, the NRC staff requested that the licensee summarize their HRA dependency analysis and confirm that there was specific justification for each minimum joint HEP used in the IEPRA below 1E-06, consistent with guidance in NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)" (Reference 37). In response to APLA RAI 6, the licensee explained that a minimum joint HEP was not applied in the IEPRA, but will be applied in the next PRA model update. The licensee clarified that HEPs below 1E-06 will only be used after a detailed review of the sequence to confirm that the timing, cues, manpower, and stress levels of the constituent human failure events justifies it. In ALPA RAI 6.b.01, the NRC staff requested that the licensee provide an implementation item requiring a joint minimum HEP of 1E-06 or a detailed review of each HEP with value lower than 1E-06. In response to ALPA RAI 15.01 (Implementation Item), the license provided the implementation item that any individual HEP less than 1E-06 will have a detailed analysis and technical justification. In the same response, the licensee proposed an associated license condition requiring completion of the implementation items before implementation of the RICT Program. The NRC staff concludes that the licensee's application of a minimum joint HEP is acceptable because the proposed HEP evaluations are consistent with NRC guidance and will be implemented prior to the RICT Program implementation.

In LAR Enclosure 2, the licensee identified 11 unresolved F&Os from the 2013 focused scope peer review. The licensee stated that each F&O will be resolved in the next model update to take place before implementation of the RICT Program at Turkey Point, and that the resolution is expected to have little effect on the calculated RICTs. In response to APLA RAI 12, the licensee stated that it had completed the F&O closure process in accordance with Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13. The F&O closure review team concluded that the licensee had adequately addressed 10 of the F&Os, but re-opened LE-D2-01 regarding electrical penetration failures that could consequently fail containment integrity. The F&O closure review team concluded that these failures would not significantly contribute to LERF, but found no documentation in the licensee's PRA addressing the issue. The NRC staff notes that the original F&O indicated that such failures may be important at ultimate containment failure pressure (i.e., long term overpressure failure modes). The NRC staff finds that electrical penetration failure evaluation and documentation can be adequately addressed in the future using the continuing PRA model update process, because long term overpressure containment failures are not expected to have significant impact on the large early release metric used in the RICT calculations.

In addition to the F&O closure review's re-opening of LE-D2-01, Attachment 3 to the February 15, 2018, RAI response, provided comments on 42 other F&Os for which the team concluded that the information available was not conclusive. The NRC staff has reviewed the IA team's comments and full performance level evaluations and concludes that, when the IA team was unable to identify documentation to verify that the resolution proposed by the licensee was implemented, the licensee's stated resolution is assumed to have been implemented, but not documented, and, therefore, there is no impact on the RICT due to the lack of documentation. Some other F&Os that the IA team judged to still be open involve low likelihood scenarios (e.g., neglected spray effects from ruptured piping) that are not expected to impact RICT estimates. IA team comments on three IEPRA F&Os are discussed below and one additional comment on FPRA F&Os is discussed in the FPRA section.

In the disposition of F&O HR-A2-01 found in Table 1 of Enclosure 2 to the LAR, the licensee stated that pre-initiator human failure events were always assumed to be possible, but that procedures were only evaluated for risk-significant items. The IA team noted that screening values of 3E-3 for individual and 3E-4 for multiple pre-initiator events may not capture all risk-significant events and that, contrary to the Supporting Requirement, all pre-initiators that are possible according to the procedures are not included in the screening. In its June 12, 2018, response to RAI 12.01.a and 15-01, the licensee added an implementation item to use a systematic process to review calibration procedures and practices to identify activities that if performed incorrectly, can have an adverse impact on the automatic initiation of standby safety equipment. The NRC staff finds this issue is resolved because a systematic process consistent with Supporting Requirement will be implemented.

In the disposition of F&O QU-3 found in Table 1 of Enclosure 2 to the LAR, the licensee stated that changes to four computer files that modify the PRA logic were updated and documented. The IA team noted that there was no evidence that the contents and/or the documentation for two of these files had been updated. The staff noted that improper modification of the logic models can affect the RICT calculations. In its June 12, 2018, response, the licensee reiterated that the actual changes to the logic models are appropriate and that the documentation will be enhanced. The NRC staff finds this response acceptable because the licensee has confirmed that the logic is correct and that the documentation will be enhanced.

In the disposition of IFSN-A2 and IFSN-A4-01 found in Table 1 of Enclosure 2 to the LAR, the licensee stated that that no credit for floor drains or operator actions to mitigate a flood was taken in the internal flooding models and, therefore, there was no need to credit flood alarms. The IA team comments identified a PRA document that stated that a reasonable time for the flood to be terminated was based on alarms. In its June 12, 2018, response, the licensee confirmed that no action by the operators that might curtail a flood scenario was considered: floods were allowed to persist for long periods of time (~12 to 24 hours) unless the flood source is finite, and that the documentation supporting the flooding evaluations will be updated. The NRC staff finds that no crediting operator actions to isolate floods is conservative and, therefore, that this issue is resolved.

Based on the licensee description of the review process, the NRC staff finds that the Turkey Point IEPRA is of sufficient scope and technical adequacy based on consistency with the guidance in RG 1.200, Revision 2 and after completion of the six implementation items listed in the table of implementation items in the enclosure to FPL letter L-2018-118 dated June 12, 2018. Therefore, the NRC staff finds that IEPRA is technically adequate to support the RICT Program, including RICT calculations.

Fire Probabilistic Risk Assessment

The licensee evaluated the technical adequacy of the Turkey Point FPRA model by conducting a full-scope peer review using the NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines" (Reference 38), peer review process and Part 4 (Fire PRA) of the current PRA ASME standard, ASME/ANS RA-Sa-2009, as clarified by RG 1.200. By letter dated February 15, 2018, the licensee stated that an IA team had completed the F&O closure process in accordance with Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13. The licensee did not provide the date of the F&O closure review, but Attachment 3 to the February 15, 2018, letter provided the results of the review for NRC staff to review all the F&Os. The NRC staff did not assess the implementation of the IA F&O Closure process because the F&Os were provided in its entirety.

In LAR Enclosure 2, Table 2, the licensee described how each F&O was resolved. An earlier NRC staff review of the technical adequacy of the FPRA was performed during the staff's review of the licensee's National Fire Protection Association (NFPA) 805 LAR (Reference 39). At the end of that review, the NRC staff concluded that the Turkey Point FPRA possessed sufficient technical adequacy that its quantitative results can be used to demonstrate that the change in risk due to the transition to NFPA 805 meets the acceptance guidelines in RG 1.174. The NRC staff considered the resolution of FPRA issues, as documented in the earlier review of FPRA for the NFPA 805 LAR, during its review of the FPRA to support the RICT Program.

In APLA RAI 2, the NRC staff noted that discussion, recommendation, and disposition of every FPRA F&O in Table 2, Enclosure 2 of the RICT LAR appeared to be the same as given in Table V-3 of the NFPA 805 LAR. The NRC staff's review of the NFPA 805 LAR resulted in a number of RAIs on the disposition of the FPRA F&Os in Table V-3 for the NFPA 805 application. The response to NFPA PRA RAI 29 contained in letters dated April 4, 2014 (Reference 40), and July 18, 2014 (Reference 41), summarized a variety of method and model changes that were required so that the final FPRA would only use acceptable methods. In APLA RAI 2 for the RICT LAR, the NRC staff asked if the FPRA that will be used to support the RICT calculations will be the FPRA, as modified to be acceptable for use in the NFPA 805 program.

In response to APLA RAI 2, the licensee stated that the FPRA that will be used to support the RICT calculations will be the same as was determined to be acceptable for the NFPA 805 transition and future self-approval. Except for one issue that has arisen because of new information developed by the NRC and discussed below, the NRC staff considers a FPRA accepted for NFPA 805 acceptable for the RICT Program. In APLA RAI 9, the NRC staff asked whether all the changes the FPRA required before implementation of NFPA 805 have been completed. In response to APLA RAI 9, the licensee stated that some of FPRA changes and plant modifications that are part of the NFPA 805 approval had not been completed at that time and, therefore, would not have been completed before the December 23, 2014, submittal of the RICT LAR. The NRC staff considers the apparent discrepancy of identical F&O resolutions for F&Os that should have been changed were either a timing issue or a documentation issue (the changes had not been completed and/or documented before the 2014 submittal of the RICT LAR) and finds further clarification unnecessary.

In a letter dated July 1, 2016 (Reference 42), subsequent to the approval of the Turkey Point NFPA 805 LAR, the NRC staff retired NFPA 805 Frequently Asked Question (FAQ) 08-0046 "Incipient Fire Detection Systems." This retirement reduced the PRA credit that could be taken for the Very Early Warning Fire Detection System (VEWFDS). In the letter, the NRC staff directed licensees who credited the installation of VEWFDS using the methods in FAQ 08-0046 to evaluate the impact on their PRA in accordance with their licensing bases. By letter dated November 17, 2016 (Reference 43), the NRC staff informed the industry that, "[i]f a licensee is performing a periodic or interim PRA update, performing a fire risk evaluation in support of self-approval, or submitting a future risk informed license amendment request, the staff's expectation is that they will assess the impact of new operating experience and information on their PRA analyses and incorporate the change as appropriate per Regulatory Guide 1.200, Revision 2." In December 2016, the NRC staff published new guidance on modeling VEWFDS in NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities (Delores-VEWFIRE)" (Reference 44), which replaced the retired FAQ 08-0046. In its June 12, 2018, response to APLA RAI-15.01, FPL added an implementation item confirming that VEWFDS modeling in accordance with NUREG-2180 will be incorporated into the Turkey Point FPRA model before implementation of the RICT Program. The NRC staff finds that this issue is resolved because a currently acceptable method will be implemented into the FPRA before the RICT Program is implemented.

In Attachment 3 to its February 15, 2018, RAI response, the licensee provided the IA team's comment on FPRA F&O IGN-A7. In the disposition of F&O IGN-A7 found in Table 1 of Enclosure 2 to the LAR, the licensee stated that supplemental walk downs were performed to identify any missing fire ignition sources and the analysis was updated accordingly. The IA team commented that not all ignitions sources that have been identified have been included in the FPRA. In its June 12, 2018, response to APLA RAI 12.01.d and APLA RAI 15.01, the licensee added an implementation item to include all ignition sources identified in the walkdowns in the FPRA before the RICT Program is implemented. The NRC staff finds that this issue is resolved because all the identified ignition sources method will be included in the FPRA before the RICT Program is implemented.

In Attachment 3 to the February 15, 2018, RAI response, the licensee provided the IA team's comment on FPRA F&O FSS-A1. The IA team stated that the FPRA was missing turbine generator fires. In its June 12, 2018, response to APLA RAI 15.01, the licensee added an implementation item to include new scenarios identified in the IA team disposition of FSS-A1 in the PRA. The NRC staff finds that this issue is resolved because any missing scenarios will be included in the FPRA before the RICT Program is implemented.

As a result of the review of the LAR and the supplements to the LAR, the NRC staff concludes that the licensee has either demonstrated that the methods adequately meet the Supporting Requirements in ASME/ANS-RA-Sa 2009, as clarified by RG 1.200, or the licensee will implement acceptable methods prior to RICT Program implementation, and that there are no remaining identified issues that could significantly impact the Turkey Point RICT Program. Therefore, the NRC staff finds that the FPRA will be technically adequate to support the RICT Program, including RICT calculations.

PRA Quality Conclusions

Based on the NRC staff's review of the licensee's submittal and assessments, the NRC staff concludes that the Turkey Point PRA models for internal events (including internal flooding) and fire events used to implement the RICT Program satisfy the guidance of RG 1.200. The NRC staff based this conclusion on the findings that the PRA models conform sufficiently to the applicable industry PRA standards for internal events (including internal flooding) and fires at an appropriate Capability Category, considering the acceptable disposition of the peer review and NRC staff review findings.

The NRC staff finds the quality of the PRA adequate to support the RICT Program because the licensee (1) reviewed the PRA using endorsed guidance and adequately resolved identified issues, (2) will address remaining issues through implementation items, and (3) established a periodic update and review process to update the PRA and associated CRMP model to incorporate changes made to the plant and PRA methods and data.

3.1.4.1.2 Scope of the PRA

NEI 06-09, Revision 0-A, requires a quantitative assessment of the potential impact on risk due to impacts from internal and external events, including internal fires, floods, and other significant external events. As clarified in the NRC staff's safety evaluation on NEI 06-09, Revision 0-A, other sources of risk (i.e., seismic and other external events) must be quantitatively assessed if they contribute significantly to the incremental risk of any RMTS configuration. Sources of risk shown to be insignificant contributors to configuration risk may be excluded from the RICT calculations. Additionally, shutdown risk assessment is not applicable to this LAR, because the RICT Program would only apply to Modes 1 and 2.

The licensee provided its assessment of external hazard risk for the RICT Program in LAR Enclosure 4, "Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models." According to the LAR, the licensee evaluated the following external hazards identified in NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (Reference 45):

- Aircraft impacts
- External flooding
- Extreme winds and tornadoes (including generated missiles)
- External fires
- Accidents from nearby facilities
- Pipeline accidents (e.g., natural gas)
- Release of chemicals stored at the site
- Seismic events
- Transportation accidents
- Turbine-generated missiles

Enclosure 4 in the LAR states:

> [T]he review of external hazards considers two aspects of the contribution to risk. The first is the contribution from the occurrence of beyond design basis conditions (i.e., winds greater than design). These beyond design basis conditions challenge the functionality of the systems, structures, and components (SSCs) to support safe shutdown of the plant. The second aspect addressed are the challenges caused by external conditions that are within the design basis, but still require some plant response to assure safe shutdown (i.e., high winds causing loss of offsite power). While the plant design basis assures that the safety related equipment necessary to respond to these challenges are protected, the occurrence of these conditions nevertheless cause a demand on these systems and can impact configuration risk.

In Table E4-1, the licensee reported the conclusions for each hazard that no unique PRA model for the hazard is required in order to assess configuration risk for the RICT Program. In its October 30, 2017, response to APLA RAI-17, the licensee stated that the RICT Program will include guidance to establish additional RMAs for the applicable hazards. For example, in the event of high winds, RMAs will protect the availability of the emergency diesel generators.

Based on the licensee description of an external hazard review process that considers design basis and less than design bases hazards' impact on configuration risk, and the establishment of additional RMAs for the applicable hazards, the NRC staff finds that external hazards have been evaluated and will, as appropriate, be included in the RICT Program.

3.1.4.1.3 PRA Modeling

As summarized in NEI 06-09, Revision 0-A, and the NRC staff's associated safety evaluation, a RICT for a given required action can be calculated when the specific systems or components involved are modeled directly or indirectly in the PRA. For each TS LCO required action for which the licensee proposes to apply the RICT Program, the licensee stated that: (1) the success criteria parameters used to determine PRA functional determination are the same as the design-basis success criteria parameters or, if different, plant-specific analyses used to support the PRA are justified; (2) the system is included in the PRA models, or those systems not in the PRA are addressed either in the LAR or in response to an RAI; (3) CCFs and surrogate identification are appropriately addressed; and (4) the CRMP provides the capability to select the system as out-of-service in order to calculate a RICT and that the CRMP is maintained consistent with the baseline PRA model.

Success Criteria

Table E1-1 in Enclosure 1 of the LAR, as supplemented, identifies each TS within the CRMP program and, as applicable, summarizes how the PRA success criteria differ from the design-basis success criteria. In some cases, all the design-basis success criteria are not modeled in the PRA (e.g., small LOCA hot-leg injection) or are more restrictive than the PRA success criteria (e.g., requiring only one SI pump, rather than two). Section d of the licensee's proposed TS 6.8.4.p states:

> Use of a RICT is not permitted for entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE.

This TS does not permit PRA success criteria to replace design-basis success criteria as a basis for continued operation. CTs calculated from the PRA will be based on the PRA success criteria which have been reviewed and are consistent with the PRA technical adequacy review process described in RG 1.200. Use of less-restrictive PRA success criteria solely to extend CTs when the design basis criteria can still be satisfied is consistent with NEI 06-09, Revision 0-A and the associated safety evaluation and, therefore, acceptable.

System, Surrogate, and Common Cause modelling

Table E1-1 in Enclosure 1 of the LAR, as supplemented, identifies each TS within the CRMP and, as applicable, identifies how the systems and components are implicitly or explicitly modeled in the PRA. Table E1-1 further clarifies how surrogate events can be used to include the impacts of failed systems and components not explicitly modeled in the PRA. The NRC staff requested additional information about some potentially ambiguous surrogate events, as discussed below.

In APLA RAI 3, the NRC staff requested an explanation about the event in the PRA that will be used as the bounding surrogate for the containment isolations system, LCOs 3.6.1.3, 3.6.1.7, and 3.6.4. In the letter dated June 16, 2016, the licensee clarified that the large, pre-existing containment leak surrogate event coincident with a core damage sequence leads to a large early release scenario. The NRC staff finds that the surrogate is conservative and acceptable, because all leak scenarios will contribute to the LERF metric and further differentiation on leak size is not needed.

In APLA RAI 4 and RAI 4.01, the NRC staff requested information on whether the licensee's PRA model included containment spray and emergency containment cooling systems as affecting CDF and LERF, or if they were only credited for long-term containment overpressure. In the June 16, 2017, and October 30, 2017, RAI responses, the licensee clarified that containment spray and emergency cooling systems failures impact CDF because they are modeled as providing a favorable environment for operation of some SSCs supporting hot leg injection, residual heat removal, and containment sump level instrumentation. The licensee stated that the impact on CDF and LERF can be estimated by setting the basic-event failure probabilities to true or 1.0. The NRC staff finds the inclusion of these LCOs acceptable, because the licensee stated that the PRA can model the impact of their failure on CDF.

In APLA RAI-13, the NRC staff stated that according to RG 1.177, Appendix A, Section A-1.3.1.1; "[i]f the component is down because it is being brought down for maintenance, the CCF contributions involving the component should be modified to remove the component and to only include failures of the remaining components." Accordingly, the NRC staff requested an explanation of how the CCF contribution is addressed in the PRA models and how the models are adjusted when one train of a three-train system is removed for preventive maintenance. In response to APLA RAI-13, the licensee explained that for a three-train system there would be one CCF basic event for each two-out-of-three SSC failure combination and a CCF basic event for the failure of three-out-of-three SSCs. The licensee explained that it does not adjust the contribution of CCFs for planned maintenance. Leaving all two-out-of-three and the three-out-of-three CCF basic events in the model is conservative, but this impact is offset to some extent because the two-out-of-two CCFs tend to be greater than the three-out-of-three CCFs. The NRC staff notes that the licensee's method is a straightforward simplifying calculation that has both conservative and non-conservative impacts. The NRC staff also notes that CCF probability estimates are very uncertain and retaining precision in calculations using these probabilities will not necessarily improve the accuracy of the results. Therefore, the NRC staff concludes that the licensee's method is acceptable because the calculations reasonably include CCFs after removing one train for maintenance consistent with the accuracy of the estimates.

In APLA RAI-14, NRC staff stated that according to Section A-1.3.2.1 of Appendix A of RG 1.177, when a component fails, the CCF probability for the remaining redundant components should be increased to represent the conditional failure probability due to CCF of these components, in order to account for the possibility that the first failure was caused by a CCF mechanism. In response to APLA RAI-14, the licensee proposed Paragraph e of proposed TS 6.8.4.p to account for emergent conditions where the extent-of-condition for an inoperable SSC is not complete. The proposed requirement states that the licensee will either account for the increased CCF in the RICT calculation or implement RMAs not already credited in the RICT calculation that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practical reduce the frequency of the initiating events that challenge the function(s) performed by the inoperable SSCs. The NRC staff finds that the first option is acceptable because it quantitatively incorporates the potential CCF into the estimated RICT, consistent with guidance on including CCFs in RG 1.177. The NRC staff finds the second option is acceptable because identifying the redundant and/or diverse SSCs and developing RMAs targeting the function(s) provides adequate additional confidence that the function(s) will be available while investigation into the potential for CCF is completed.

<u>CRMP Model</u>

The Turkey Point PRA model serves as the model used by the CRMP tool, which is used to perform the RICT calculations. The CRMP tool models change in CDF and LERF above the zero-maintenance baseline plant configuration, and converts to average annual values. In order to translate the baseline Turkey Point PRA model for use in the CRMP model, adjustments must be made to the baseline PRA model. These adjustments are described in the LAR Enclosure 8.

The CRMP tool used to perform the RICT calculations provides a user interface that supports the RICT Program by providing a method to evaluate the plant configuration.

In APLA RAI 07, the NRC staff requested the licensee to provide a discussion of the changes made to the baseline PRA model to produce the CRMP model and how it is assured that these changes are appropriate and comprehensive. In response to APLA RAI 07, the licensee explained that the baseline model is configured by removing mutually exclusive maintenance events logic, and altering flag file and alignment events to allow the risk monitoring software to perform configuration-specific risk analyses.

In APLA RAI-18, the NRC staff requested the licensee to provide a discussion of how the two units will calculate and apply a RICT to both units, simultaneously. In response to APLA RAI-18, the licensee explained that there is one fault-tree model comprised of two complete, interdependent PRA models (one for each unit). The licensee also stated that the units have several cross-ties and shared systems and components that are modeled once, with proper ties to the individual units' fault tree logic. In this manner, the risk-monitoring software managing the RICT calculations can properly calculate the impact of a configuration regardless of whether it impacts one or both units.

The NRC staff reviewed the licensee's information and concluded that the scope of SSCs to which the RICT Program are applied are appropriately included in the PRA models and in the CRMP. Therefore, the NRC staff finds that the licensee's PRA modeling is consistent with NEI 06-09, Revision 0-A, guidance subject to the conditions in Section 4.0 of the NRC staff's associated safety evaluation.

PRA Modelling Conclusions

The NRC staff reviewed the licensee's information and concludes that the PRA modeling used to support the RICT Program is able to appropriately model alignments of components during periods when the RICT will be calculated. Therefore, the NRC staff finds that the licensee has satisfied the intent of RG 1.177 (Section 2.3.3), and RG 1.174 (Section 2.3), and that the PRA modeling is appropriate for this application.

3.1.4.1.4 Assumptions

Using PRAs to evaluate TS changes requires consideration of a number of assumptions made within the PRA that can have a significant influence on the ultimate acceptability of the proposed changes. With regard to changes to CTs, the following assumptions were evaluated:

Enclosure 9 of the LAR states that the detailed process of identifying, characterizing the key assumptions and qualitative screening of model uncertainties is found in Section 5.3 (Section 7.1 in Revision 1) of NUREG-1855. NUREG-1855 references Electric Power Research Institute report, TR 1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments" (Reference 46), which provides specific methods for identifying key sources of model uncertainty within the context of the significant contributors to the various risk metrics that are relevant to a particular application.

LAR Table E9-1 includes potential key assumptions, some of which were identified in the peer reviews, and additional plant-specific and generic key assumptions that were identified by the licensee in a 2014 Uncertainty Notebook, which is part of the PRA documentation. The licensee briefly discussed and then dispositioned the impact of each identified key assumption on the

RICT Program. Two assumptions regarding unmodeled events—human induced initiating events and CCFs for electrical buses and panels—were determined to be unimportant because their impact would be mitigated by RMAs. The licensee identified the remaining assumptions as slightly conservative, neutral, or slightly non-conservative assumptions; but, for each assumption concluded that they do not significantly affect the RICT calculation and that no additional considerations are required.

Based on the identification and disposition of the significant PRA assumptions described above, the NRC staff finds that the licensee has satisfied the intent of applicable parts of RG 1.177 (Section 2.3.4), and that the assumptions for risk evaluation of extended CTs are appropriate for this application.

3.1.4.1.5 Sensitivity and Uncertainty Analyses

Risk-informed analyses of TS changes can be affected by uncertainties regarding the assumptions made during the PRA model's development and application. Typically, the risk resulting from TS CT changes is relatively insensitive to most uncertainties, because the uncertainties tend to affect similarly both the base case and the changed case. 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. The licensee discussed sources of uncertainty and assumptions in Enclosure 9 of the LAR.

Based on its conclusions that sources of uncertainty identified in Enclosure 9 are included in the RMTS or do not significantly affect the RICT calculations, the licensee performed no sensitivity analyses for key assumptions. The licensee reported the sensitivity studies it performed to support the closure of F&Os without any modifications to the PRA including (1) increasing internal flooding event frequency to address the lack of human caused flooding events and (2) modifying transient fire frequencies. The licensee reported very small impacts and concluded that changes to the PRA were not needed.

Based on the above reported studies and its conclusions that sources of uncertainty identified in Enclosure 9 are included in the RMTS or do not significantly affect the RICT calculations, the NRC staff finds that the licensee's evaluation is consistent with the guidance in NEI 06-09, Revision 0-A, and is, therefore, acceptable.

3.1.4.1.6 PRA Results and Insights

The proposed change implements a process to determine RICTs for TSs, rather than specific changes to individual TS CTs. NEI 06-09, Revision 0-A, requires periodic assessment of the risk incurred due to operation beyond the front-stop CTs due to implementation of a RICT Program, and comparison to the guidance of RG 1.174, for small increases in risk. As with other unique, risk-informed applications, supplemental risk acceptance guidelines that complement the RG 1.174 guidance are appropriate.

Further, NEI 06-09, Revision 0-A, requires that configuration risk be assessed to determine the RICT, and establishes the criteria for ICDP and ILERP on which to base the RICT. An ICDP of 1E-5 and an ILERP of 1E-6 are used as the risk measures for calculating individual RICTs. These limits are consistent with NUMARC 93-01, Revision 4A. The use of these limits in NEI 06-09, Revision 0-A, aligns the TS CTs with the risk management guidance used to support plant programs for the Maintenance Rule, and the NRC staff accepted these supplemental risk acceptance guidelines for RMTS programs in its approval of NEI 06-09, Revision 0-A.

NEI 06-09, Revision 0-A, as modified by the limitations and conditions in the NRC staff's safety evaluation, requires that the cumulative impact of implementation of an RMTS be periodically assessed and: (1) shown to result in a total risk impact below 1E-5/year for changes to CDF, (2) shown to result in a total risk impact below 1E-6/year for changes to LERF, and (3) the total CDF and total LERF must be reasonably shown to be less than 1E-4/year and 1E-5/year, respectively. The licensee indicated in the October 30, 2017, response to APLA RAI-02.01, that the estimated total CDF and LERF are 1.91E-4/year and 1.81E-4/year CDF for Units 3 and 4 respectively, and 1.75E-5/year and 1.80E-5/ year LERF for Units 3 and 4, respectively. These values exceed the criteria of NEI 06-09, Revision 0-A. In its October 30, 2017, RAI response, the licensee identified several new FPRA methods in NUREGs and FAQs that it plans to implement, which will decrease the fire-related CDF and LERF and, therefore, the total values.

In the June 12, 2018, response to APLA RAI 15.01, the licensee provided an implementation item to "[c]onfirm that the all hazards CDF and LERF estimates achieved using NRC accepted methods will be less than 1E-04 per year and 1E-05 per year, respectively[,]" prior to RICT Program implementation.

The NRC staff finds that the licensee has identified some conservatisms in its FPRA that, when replaced, will reduce the total CDF and LERF; that other changes to the PRA methods to the PRA will only implement PRA methods approved by the staff; and that the implementation of the items referenced in the proposed license condition discussed in Section 4.0 of this safety evaluation provides reasonable assurance that the CDF and LERF guidelines will be met before the licensee implements the RICT Program.

3.1.4.1.7 Implementation of the RICT Program

Because NEI 06-09, Revision 0-A, involves the real-time application of PRA results and insights by a licensee, the NRC staff reviewed the licensee's description of programs and procedures associated with implementation of the RICT Program in Attachment 1 (and its enclosures) of the LAR. The administrative controls on the PRA and changes to the PRA provide confidence that the PRA results are reasonable, and the administrative controls on the plant personnel using the RICT should provide confidence that the RICT Program will be appropriately applied.

The quality assurance practices for the PRA models include meeting the ASME/ANS-RA-Sa 2009 PRA standards and RG 1.200, which includes guidance for performing peer reviews and focused scope peer reviews. The quality assurance practices for the PRA models are discussed by the licensee in Enclosures 2 and 7 of the LAR. According to Enclosure 8 of the LAR, future changes made to the baseline PRA model, changes made to the baseline PRA model for translation to the online model, and changes made to the online model configuration files, are controlled and documented by plant procedures.

Future changes to the PRA are expected over time to reflect changes in PRA methods, and changes to the as-built, as-operated, and maintained plant to reflect the operating experience at the plant, as specified in RG 1.200. Changes in PRA methods are addressed by the following proposed license condition in the Attachment Markup of the Unit 3 and Unit 4 Operating Licenses, in the licensee's letter dated June 12, 2018:

> The risk assessment approach and methods, shall be acceptable to the NRC, be based on the as-built, as-operated, and maintained plant, and reflect the operating experience of the plant as specified in RG 1.200. Methods to assess

> the risk from extending the completion times must be PRA methods accepted as part of this license amendment, or other methods approved by the NRC for generic use. If the licensee wishes to change its methods, and the change is outside the bounds of this license condition, the licensee will seek prior NRC approval via a license amendment.

The NRC staff finds that this license condition is acceptable because it assures that the RICT Program will be adequately implemented using models, methods, and approaches that are consistent with applicable guidance and acceptable to the NRC.

Changes to the as-built, as-operated, and maintained plant to reflect the operating experience at the plant are discussed in LAR Enclosure 7. Enclosure 7 summarizes the PRA configuration-control process; delineates the responsibilities and guidelines for updating the full power internal event, internal flood, fire, and seismic PRA models; and includes both periodic and interim PRA model updates. The licensee stated that the process includes provisions for monitoring potential impact areas affecting the technical elements of the PRA models (e.g., due to plant changes, plant/industry operational experience, or errors or limitations identified in the model), assessing the individual and cumulative risk impact of unincorporated changes, and controlling the model and necessary computer files, including those associated with the CRMP model.

In its June 16, 2016, response to APLA RAI 7, the licensee summarized the changes that need to be made to the baseline PRA to generate the CRMP model. The baseline PRA model is modified by removing mutually exclusive maintenance events logic excluding configurations prohibited by plant procedures or guidelines, and altering the flag file and alignment events to allow those using the risk monitoring software for a configuration-specific risk analysis to designate the alignments and configuration in effect at the time. The licensee stated in Enclosure 8 of the LAR that the plant procedures specify that an acceptance test is performed after every CRMP model update. This test verifies proper translation of the baseline PRA models and acceptance of all changes made to the baseline PRA models into the CRMP model. This test also verifies correct mapping of plant components for the basic events in the CRMP model. The NRC staff concludes that the CRMP model used to calculate the RICTs is acceptable because the underlying PRA models will remain acceptable, the transition between the baseline and the CRMP model is reasonable, and the acceptance test should verify that the CRMP model is consistent with the underlying baseline PRA.

As described in Enclosure 10 of the LAR, the licensee has qualification and training programs for development, maintenance, and use of the CRMP model. The licensee identifies the attributes that the RICT Program procedures will address consistent with NEI 06-09, Revision 0-A. The licensee also identified the plant personnel that will be trained and the different types of training that the different plant personnel receive. This includes training for individuals who will be directly involved in the implementation of the RICT Program, as well as other individuals who may have some involvement with the RICT Program.

The NRC staff finds that the program described in Enclosure 10 will establish appropriate programmatic and procedural controls for the licensee's RICT Program, consistent with the guidance of NEI 06-09, Revision 0-A. Training of plant personnel shall be provided throughout all levels of the organization, commensurate with each position's responsibilities within the RICT Program, as described in NEI 06-09, Revision 0-A. The NRC staff notes that the licensed operators in the control room have responsibility for assuring compliance with the TSs, and that the RICT Program training provided assures the licensee's staff understands risk concepts, and provides them with the necessary skills to determine the appropriate RICT when operating under an extended CT within the RICT Program.

The LAR, as supplemented, summarizes the administrative controls used to support implementation of the RICT Program including maintenance of the PRA models used by the program and the training for plant personnel throughout all levels of the organization. Therefore, the NRC staff finds that the licensee has appropriate administrative controls in place to assure proper implementation of the RICT Program.

3.1.4.2 Tier 2: Avoidance of Risk-Significant Plant Configurations

Tier 2 provides that a licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is taken out-of-service in accordance with the proposed TS change.

NEI 06-09, Revision 0-A, does not permit voluntary entry into high-risk configurations, which would exceed instantaneous CDF and LERF limits of 1E-3/year and 1E-4/year, respectively. It further requires implementation of RMAs when the actual or anticipated risk accumulation during a RICT will exceed one-tenth of the ICDP or ILERP limit. Such RMAs may include rescheduling planned activities to lower risk periods or implementing risk-reduction measures. The limits established for entry into a RICT and for RMA implementation are consistent with the guidance of NUMARC 93-01, Revision 4A, endorsed by RG 1.160, Revision 3, as applicable to plant maintenance activities. The RICT Program requirements and criteria are consistent with the principle of Tier 2 to avoid risk-significant configurations.

Based on the licensee's incorporation of NEI 06-09, Revision 0-A, in TS 6.8.4.p, and because the proposed risk acceptance guidelines are consistent with the guidance of RG 1.174 and RG 1.177, the NRC staff finds the licensee's Tier 2 program is acceptable and supports the proposed implementation of the RICT Program.

3.1.4.3 Tier 3: Risk-Informed Configuration Risk Management

The third tier provides 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.

NEI 06-09, Revision 0-A, addresses Tier 3 guidance by requiring assessment of the RICT to be based on the plant configuration of all SSCs that might impact the RICT, including safety-related and non-safety-related SSCs. If a risk-significant plant configuration exists, based on the expectation of exceeding a threshold of one-tenth of the risk on which the RICT is based, then compensatory measures and RMAs are required to be implemented. Thus, the RICT Program provides a methodology to assess and address risk-significant configurations. Further, reassessment of any plant configuration changes is also required to be completed in a timely manner, based on the more restrictive limit of any applicable TS action requirement or a maximum of 12 hours after the configuration change occurs.

Based on the licensee's incorporation of NEI 06-09, Revision 0-A, in TS 6.8.4.p, and because the proposed changes are consistent with the Tier 3 guidance of RG 1.177, Revision 1, the NRC staff finds that the proposed changes are acceptable.

3.1.4.4 Key Principle 4 Conclusions

The licensee has demonstrated the technical adequacy and scope of its PRA models, and that with the implementation items, the models can support implementation of the RICT Program for determining CTs. Consideration of key assumptions and sources of uncertainty consistent with the guidance with NEI 06-09, Revision 0-A, has been made. The risk metrics are consistent with the approved methodology of NEI 06-09, Revision 0-A, and the RICT Program is controlled administratively through plant procedures and training. The RICT Program follows the NRC-approved methodology in NEI 06-09, Revision 0-A. The NRC staff concludes that the RICT Program satisfies the fourth key safety principle of RG 1.177 and is, therefore, acceptable.

3.1.5 Key Principle 5: Performance Measurement Strategies – Implementation and Monitoring Program

Regulatory Guides 1.174 and 1.177 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. Two performance monitoring strategies are identified in the licensee's LAR: Monitoring of the total cumulative impact of extending the CTs, and monitoring the reliability and availability of SSCs impacted by the RICT Program.

Section 3.3.3 of NEI 06-09, Revision 0-A, requires that the licensee track the risk associated with all entries beyond the front stop CT, and Section 2.3.1 provides a requirement for assessing cumulative risk, including a periodic evaluation of any increase in risk due to the use of the RMTS program to extend the CTs. According to Enclosure 12 of the LAR, the licensee calculates cumulative risk at least every refueling cycle, not to exceed 24 months, which is consistent with NEI 06-09, Revision 0-A. The licensee converts the cumulative ICDP and the ILERP into average annual values that are then compared to the acceptable risk increase guidelines of RG 1.174. If any limits are exceeded, corrective actions are taken to ensure future plant operational risk is within the acceptance guidance. This evaluation assures that RMTS program implementation meets RG 1.174 guidance for small risk increases. The licensee is implementing NEI 06-09, Revision 0-A, via the RICT Program and, therefore, complies with this RMTS program.

An implementation and monitoring program is also intended to ensure that any changes to the reliability and availability of SSCs within the scope of the RICT Program be identified and incorporated into the RICT models. Regulatory Guide 1.174 states that monitoring performed in conformance with the Maintenance Rule, 10 CFR 50.65, can be used when the monitoring performed is sufficient for the SSCs affected by the risk-informed application. Enclosure 11 of the LAR states that the SSCs in the scope of the RICT Program are also in the scope of the Maintenance Rule and, therefore, the Maintenance Rule monitoring can be used.

The NRC staff concludes that the RICT Program satisfies the fifth key safety principle of RG 1.177, Revision 1, and RG 1.174 by, in part, monitoring the average annual cumulative risk increase as described in NEI 06-09, Revision 0-A, and using this average annual increase to ensure that the program as implemented meets RG 1.174 guidance for small risk increases; this aspect of the RICT Program is, therefore, acceptable. Additionally, the NRC staff concludes that the RICT Program satisfies the fifth key safety principle of RG 1.177, Revision 1, and RG 1.174 because, in part, all the affected SSCs are within the Maintenance Rule program that can be used to monitor changes to the reliability and availability of these SSCs.

3.2 Evaluation of Other Proposed TS Changes

3.2.1 Revision of ACTIONs in TS 3/4.7.1.5

The licensee proposed to revise the ACTIONs associated with LCO 3.7.1.5. The LCO requires that each MSIV be operable in MODE 1 (Power Operation), MODE 2 (Startup) and MODE 3 (Hot Standby). The existing ACTIONs required:

MODE 1:

> With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 24 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

> With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

If an MSIV became inoperable during MODE 1 operation, the applicable ACTION would direct the licensee to restore the valve within 24 hours or place unit in MODE 3 within the following 6 hours. However, in shutting down the unit as required by the ACTION, the unit would enter MODE 2. Once the unit entered Mode 2, the second ACTION would become applicable. The second action would allow a continuation of operation in MODE 2, provided the inoperable valve is closed. As a result, the unit would not have to continue the shutdown to MODE 3, as originally required by the first ACTION.

The licensee proposed changes to the ACTION applicable in MODE 1 to correct this inconsistency between the two ACTIONS. The proposed change would revise the ACTION applicable in MODE 1 to require the unit to be placed in MODE 2 within 6 hours, if the inoperable MSIV is not restored to operable status within 24 hours. The NRC staff reviewed this proposed change and determined that this change is necessary to correct the inconsistency between the ACTIONs.

The licensee also proposed changes to the ACTION applicable in MODES 2 and 3: The revised ACTION would state:

> With one or more MSIVs inoperable, subsequent operation in MODE 2 or 3 may continue provided:
>
> 1. The inoperable MSIVs are closed within 8 hours, and
>
> 2. The inoperable MSIVs are verified closed once per 7 days.
>
> Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

The proposed ACTION would expand the applicability to allow the inoperability of multiple MSIVs, and to specify the time by which the MSIV is required to be closed and also specify a periodic re-verification that the valve is properly closed. The NRC staff reviewed the proposed change and considered the fact that the valves in the closed position are satisfying the assumptions made in the safety analysis. The periodic verification that the inoperable valves are correctly positioned provides assurance that the assumptions in the safety analysis continue to be satisfied. Further, the NRC staff finds that the proposed changes to LCO 3.7.1.5 are consistent with Standard Technical Specification 3.7.2, "Main Steam Isolation Valves (MSIVs)," in NUREG-1431.

For the reasons described above, the NRC staff finds that the proposed changes meet the requirements of 10 CFR 50.36(c)(2) and are acceptable.

*3.2.2 Deletion of Footnote ** in TS 3/4.6.2.1*

Footnote **, applicable to LCO 3.6.2.1, ACTION a, allowed for a one-time extension of the CT to restore an inoperable containment spray system from 72 hours to 14 days. The purpose of the one-time extension was to allow for the repair of the 3A containment spray pump, which has since been completed. As this provision is no longer applicable, the licensee proposed to delete footnote **.

The NRC staff finds that the proposed change is administrative in nature, and is acceptable.

3.2.3 Typographical Corrections

The licensee proposed to make the following changes to correct typographical errors:

- Replace "control room" with "Control Room in Table 3.3-2, ACTION 24
- Replace "as" with "at" in TS 3/4.5.2, ACTION g
- Replace "oppsite" with "opposite" in TS 3.8.2.1, ACTION b

The NRC staff finds that the proposed changes are editorial and administrative in nature, and are acceptable.

4.0 CHANGES TO THE OPERATING LICENSE

In a letter dated February 15, 2018, the licensee proposed the following changes to the Turkey Point 3 and 4 operating licenses:

I. FPL is authorized to implement the RICT Program as approved in License Amendment No. XXX subject to the following conditions:

   1. FPL will complete the items listed in the table of implementation items in the enclosure to FPL letter L-2018-118 dated June 12, 2018, prior to implementation of the RICT Program.

   2. The risk assessment approach and methods, shall be acceptable to the NRC, be based on the as-built, as-operated, and maintained plant, and reflect the operating experience of the plant as specified in RG 1.200. Methods to assess the risk from extending the completion times must be PRA methods accepted as part of this license amendment, or other methods approved by the NRC for generic use. If the

> licensee wishes to change its methods, and the change is outside the bounds of this license condition, the licensee will seek prior NRC approval via a license amendment.

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 6.8.4.p, and the implementation items in FPL letter L-2018-118. Prior NRC approval will also be required for changes to the PRA methods that have not been previously approved by the NRC in this safety evaluation or methods approved for generic use.

In the LAR and in the licensee's responses to NRC staff's RAIs, there were certain specific actions that the NRC staff identified as being necessary to support the conclusion that the implementation of the proposed program met the requirements of the RICT Program. Accordingly, the NRC staff's finding on the acceptability of implementation of the RICT Program for the TS LCOs in this safety evaluation is dependent on the completion of the implementation items identified in the licensee's June 12, 2018, letter and listed below:

Item	Implementation Date
1. The VEWFDS modeling in accordance with NUREG-2180 will be incorporated into the Turkey Point PRA model.	Prior to implementation of the RICT Program
2. FPL will use a systematic process to review calibration procedures and practices to identify activities that, if performed incorrectly, can have an adverse impact on the automatic initiation of standby safety equipment. (See RAI 12.01a.)	
3. Ignition sources identified in finding 1-18 and subsequent extent of condition walkdowns will be included in the fire PRA. (See RAI 12.01d.)	
4. New scenarios identified as part of finding 10-20 to address Turbine Generator fires will be included in the fire PRA.	
5. Confirm that the all hazards CDF and LERF estimates achieved using NRC accepted methods will be less than 1E-04 per year and 1E-05 per year, respectively.	
6. Implement a joint HEP floor of 1E-06 in the internal events model. For future model updates, once the human failure event (HFE) combinations have been analyzed and the HEP floor of 1E-06 applied, individual HFE combination probabilities may be set below 1E-06 if a detailed analysis is performed and technical justification is provided.	

The NRC staff finds that the licensee's proposed incorporation of these measures in a license condition, and the implementation of these items prior to implementation of the RICT Program is appropriate because they assure adequate implementation of the RICT Program using models, methods, and approaches consistent with applicable guidance that are acceptable to the NRC. For each implementation item, the licensee and the NRC staff have reached a satisfactory resolution involving the level of detail and main attributes that will be incorporated into the program upon completion. The NRC staff, through an onsite audit or during future inspections, may examine the closure of the implementation items, with the expectation that any issues discovered during this review, or concerns with regard to adequate completion of the implementation item, would be tracked and dispositioned appropriately under the licensee's corrective action program and subject to appropriate NRC enforcement action.

5.0 SUMMARY

5.1 NRC Staff Findings and Conclusions

The NRC staff finds that the licensee's proposed implementation of the RICT Program for the identified scope of required actions is consistent with the guidance of NEI 06-09, Revision 0-A, subject to the limitations and conditions evaluated in Section 4.0 of this safety evaluation. The licensee's methodology for assessing the risk impact of extended CTs, including the individual CT extension impacts in terms of ICDP and ILERP, and the overall program impact in terms of ΔCDF and ΔLERF, is accomplished using PRA models for internal events including internal fires and floods of sufficient scope and technical adequacy based on consistency with the guidance of RG 1.200 and completion of the six implementation items listed in the table of implementation items in the enclosure to FPL letter L-2018-118, dated June 12, 2018, and listed above. For external hazards that do not have PRA models, the licensee has determined that no unique PRA model for the hazard is required in order to assess configuration risk for the RICT Program and stated that the RICT Program will include guidance to establish additional RMAs for the applicable hazards.

The RICT calculation uses the PRA model, as translated into the CRMP tool, and the licensee has an acceptable process in place to ensure the quality of the translation. In addition, the NRC staff finds that the proposed implementation of the RICT Program addresses the RG 1.177 defense-in-depth philosophy and safety margins to ensure that the safety margins are adequately maintained, and includes adequate administrative controls as well as performance monitoring programs.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the State of Florida official (Ms. Cynthia Becker, M.P.H., Chief of the Bureau of Radiation Control, Florida Department of Health) on October 24, 2018 (ADAMS Accession No. ML18303A243), of the proposed issuance of the amendments. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the use of facility components located within the restricted area as defined in 10 CFR Part 20 or a surveillance requirement. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The

Commission has previously issued a proposed finding, which was published in the FR on August 29, 2017 (82 FR 41069), that the amendments involve NSHC, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

8.0 CONCLUSION

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

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Date: December 3, 2018

SUBJECT: TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4 - ISSUANCE OF AMENDMENTS REGARDING ADOPTION OF RISK-INFORMED COMPLETION TIMES IN TECHNICAL SPECIFICATIONS (CAC NOS. MF5455 AND MF5456; EPID L-2014-LLA-0002) DATED DECEMBER 3, 2018

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ADAMS Accession No.: ML18270A429 *by memorandum **by e-mail

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DATE	10/24/2018	10/23/2018	9/4/2018	8/17/2018
OFFICE	NRR/DE/EEOB/BC*	NRR/DRA/APLA/BC*	NRR/DSS/SCPB/BC**	NRR/DSS/SRXB/BC**
NAME	JQuichocho	SRosenberg	SAnderson	JWhitman (DWoodyatt for)
DATE	6/29/2018	7/25/2018	10/25/2018	10/19/2018
OFFICE	NRR/DSS/STSB/BC**	OGC (NLO w/edits)	NRR/DORL/LPL2-2/BC	NRR/DORL/LPL2-2/PM
NAME	VCusumano	STurk	UShoop	MWentzel
DATE	10/24/2018	11/7/2018	11/29/2018	12/3/2018

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