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

February 12, 2025

Robert Coffey
Executive Vice President, Nuclear
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 AMENDMENT NOS. 301 AND 294 REGARDING INCORPORATION OF ADVANCED FUEL PRODUCTS AND EXTENSION OF SURVEILLANCE INTERVALS TO FACILITATE TRANSITION TO A 24-MONTH FUEL CYCLE (EPID L-2023-LLA-0161)

Dear Robert Coffey:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 301 to Subsequent Renewed Facility Operating License No. DPR-31 and Amendment No. 294 to Subsequent Renewed Facility Operating License No. DPR-41 for Turkey Point Nuclear Generating (Turkey Point), Unit Nos. 3 and 4, respectively. These amendments revise Turkey Point's licensing basis by incorporating advanced fuel features (i.e., AXIOM® cladding, ADOPT™ fuel pellets, and a PRIME™ fuel skeleton), extend technical specifications (TS) surveillance intervals, modify TS Allowable Values, apply an updated instrument channel setpoint uncertainty evaluation methodology, and make conforming changes to the Updated Final Safety Analysis Report to facilitate a transition to 24-month fuel cycles. The amendments are in response to your application dated November 15, 2023, as supplemented by letters dated February 9, October 3, October 31, and November 12, 2024.

Enclosure 3 to this letter contains Proprietary Information. When separated from Enclosure 3, this letter is DECONTROLLED.

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

Sincerely,

/RA/

Michael Mahoney, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosures:
1. Amendment No. 301 to DPR-31
2. Amendment No. 294 to DPR-41
3. Safety Evaluation (Proprietary)
4. Safety Evaluation (Non-Proprietary)

cc: Listserv

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING, UNIT NO. 3

AMENDMENT TO SUBSEQUENT RENEWED FACILITY OPERATING LICENSE

Amendment No. 301
Subsequent 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 November 15, 2023, as supplemented by letters dated February 9, October 3, October 31, and November 12, 2024, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Subsequent 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. 301, are hereby incorporated into this subsequent renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this subsequent renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license is also amended to authorize revision to the Updated Final Safety Analysis Report, as set forth in the licensee's application dated November 15, 2023, as supplemented by letters dated February 9, October 3, October 31, and November 12, 2024, and evaluated in the NRC staff's safety evaluation enclosed with this amendment.

4. This license amendment is effective as of the date of its issuance and shall be implemented no later than the Unit No. 3 spring 2026 reload campaign.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: February 12, 2025

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 SUBSEQUENT RENEWED FACILITY OPERATING LICENSE

Amendment No. 294
Subsequent 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 November 15, 2023, as supplemented by letters dated February 9, October 3, October 31, and November 12, 2024, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Subsequent 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. 294 are hereby incorporated into this subsequent renewed operating license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this subsequent renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license is also amended to authorize revision to the Updated Final Safety Analysis Report, as set forth in the licensee's application dated November 15, 2023, as supplemented by letters dated February 9, October 3, October 31, and November 12, 2024, and evaluated in the NRC staff's safety evaluation enclosed with this amendment.

4. This license amendment is effective as of the date of its issuance and shall be implemented no later than the Unit No. 4 spring 2025 reload campaign.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: February 12, 2025

ATTACHMENT TO LICENSE AMENDMENT NOS. 301 AND 294

TURKEY POINT NUCLEAR GENERATING, UNIT NOS. 3 AND 4

SUBSEQUENT RENEWED FACILITY OPERATING LICENSE NOS. DPR-31 AND DPR-41

DOCKET NOS. 50-250 AND 50-251

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

Remove	Insert
DPR-31, page 3	DPR-31, page 3
DPR-41, page 3	DPR-41, page 3

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

Remove	Insert
3.3.1-11	3.3.1-11
3.3.1-12	3.3.1-12
3.3.1-13	3.3.1-13
3.3.1-14	3.3.1-14
3.3.2-7	3.3.2-7
3.3.2-8	3.3.2-8
3.3.2-9	3.3.2-9
3.3.2-10	3.3.2-10
3.3.2-11	3.3.2-11
3.3.2-12	3.3.2-12
3.3.6-4	3.3.6-4
3.3.6-5	3.3.6-5
3.6.3-5	3.6.3-5
4.0-1	4.0-1
5.5-13	5.5-13
5.6-3	5.6-3
5.6-4	5.6-4
5.6-5	5.6-5
5.6-6	-------

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. 301, are hereby incorporated into this subsequent renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this subsequent renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Deleted

D. Fire Protection

FPL shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment requests dated June 28, 2012 and October 17, 2018 (and supplements dated September 19, 2012; March 18, April 16, and May 15, 2013; January 7, April 4, June 6, July 18, September 12, November 5, and December 2, 2014; and February 18, 2015; October 24, and December 3, 2018; and January 31, 2019), and as approved in the safety evaluations dated May 28, 2015 and March 27, 2019. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the

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. 294, are hereby incorporated into this subsequent renewed operating license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this subsequent renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Deleted.

D. Fire Protection

FPL shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment requests dated June 28, 2012 and October 17, 2018 (and supplements dated September 19, 2012; March 18, April 16, and May 15, 2013; January 7, April 4, June 6, July 18, September 12, November 5, and December 2, 2014; and February 18, 2015; October 24, and December 3, 2018; and January 31, 2019), and as approved in the safety evaluations dated May 28, 2015 and March 27, 2019. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the

Table 3.3.1-1 (page 1 of 10)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.12	NA	NA
	3(a),4(a),5(a)	2	C	SR 3.3.1.12	NA	NA
2. Power Range Neutron Flux						
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7(b)(c) SR 3.3.1.9(b)(c)	≤ 108.6% RTP	108.0% RTP
b. Low	1(d),2	4	E	SR 3.3.1.1 SR 3.3.1.8(b)(c) SR 3.3.1.9(b)(c)	≤ 25.6% RTP	≤ 25% RTP
3. Intermediate Range Neutron Flux	1(d),2	2	F, G	SR 3.3.1.1 SR 3.3.1.8(b)(c) SR 3.3.1.9(b)(c)	≤ 25.6% RTP	≤ 25% RTP
4. Source Range Neutron Flux	2(e)	2	H, I	SR 3.3.1.1 SR 3.3.1.8(b)(c) SR 3.3.1.9(b)(c)	≤ 1.05 E5 cps	≤ 1.0 E5 cps
	3(a),4(a),5(a)	2	I, J	SR 3.3.1.1 SR 3.3.1.7(b)(c) SR 3.3.1.9(b)(c)	≤ 1.05 E5 cps	≤ 1.0 E5 cps

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully insert.

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodology used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2.

(d) Below the P-10 (Power Range Neutron Flux) interlocks.

(e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(f) Deleted

(g) Deleted

Table 3.3.1-1 (page 2 of 10)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
5. Overtemperature ΔT	1,2	3	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7(b)(c) SR 3.3.1.10(b)(c)	Refer to Note 2 (Page 3.3.1-18)	Refer to Note 1 (Page 3.3.1-17)
6. Overpower ΔT	1,2	3	E	SR 3.3.1.1 SR 3.3.1.7(b)(c) SR 3.3.1.10(b)(c)	Refer to Note 4 (Page 3.3.1-20)	Refer to Note 3 (Page 3.3.1-19)
7. Pressurizer Pressure - Low	1(h)	3	L	SR 3.3.1.1 SR 3.3.1.7(b)(c) SR 3.3.1.10(b)(c)	≥ 1830 psig	≥ 1835 psig
8. Pressurizer Pressure - High	1,2	3	L	SR 3.3.1.1 SR 3.3.1.7(b)(c) SR 3.3.1.10(b)(c)	≤ 2390 psig	≤ 2385 psig
9. Pressurizer Water Level - High	1(h)	3	L	SR 3.3.1.1 SR 3.3.1.7(b)(c) SR 3.3.1.10(b)(c)	≤ 92.2%	≤ 92.0%
10. Reactor Coolant Flow - Low						
a. Single Loop	1(i)	3 per loop	L	SR 3.3.1.1 SR 3.3.1.7(b)(c) SR 3.3.1.10(b)(c)	≥ 89.6%	90.0%

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodology used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2.

(f) Deleted

(g) Deleted

(h) Above the P-7 (Low Power Reactor Trips Block) interlock.

(i) Above the P-8 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 3 of 10)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
10. Reactor Coolant Flow – Low (continued)						
b. Two Loops	1(j)	3 per loop	L	SR 3.3.1.1 SR 3.3.1.7(b)(c) SR 3.3.1.10(b)(c)	≥ 89.6%	90.0%
11. Steam Generator (SG) Water Level – Low Low	1,2	3 per SG	L	SR 3.3.1.1 SR 3.3.1.7(b)(c) SR 3.3.1.10(b)(c)	≥ 15.5%	16.0%
12. SG Water Level – Low	1,2	2 per SG	L	SR 3.3.1.1 SR 3.3.1.7(b)(c) SR 3.3.1.10(b)(c)	≥ 15.5%	16.0%
Coincident with Steam Flow/Feedwater Flow Mismatch	1,2	2 per SG	L	SR 3.3.1.1 SR 3.3.1.7(b)(c) SR 3.3.1.10(b)(c)	≤ 20.7% below rated steam flow	20.0% below rated steam flow
13. Undervoltage – 4.16 kV Buses A and B	1(h)	2 per bus	L	SR 3.3.1.10(b)(c)	≥ 69% bus voltage	≥ 70% bus voltage
14 Underfrequency RCPs Breakers Open	1(h)	2 per bus	N	SR 3.3.1.10(b)(c)	≥ 56.08 Hz	≥ 56.1 Hz

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodology used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2.

(f) Deleted

(g) Deleted

(h) Above the P-7 (Low Power Reactor Trips Block) interlock.

(j) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range neutron Flux) interlock.

Table 3.3.1-1 (page 4 of 10)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
15. Turbine Trip						
a. Emergency Trip Header Pressure	1[(h)]	3	L	SR 3.3.1.10[(b)(c)] SR 3.3.1.13	≥ 901 psig	1000 psig
b. Turbine Stop Valve Closure	1[(h)]	2	L	SR 3.3.1.10 SR 3.3.1.13	Fully Closed	Fully Closed
16. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	O	SR 3.3.1.12	NA	NA
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2[(e)]	2	P	SR 3.3.1.9[(b)(c)] SR 3.3.1.11[(b)(c)]	≥ 6E-11 amp	Nominal 1E-10 amp
b. Low Power Reactor Trips Block, P-7						
1) P10 Input	1	4	Q	SR 3.3.1.9[(b)(c)] SR 3.3.1.11[(b)(c)]	≤ 13% RTP	Nominal 10% RTP
2) Turbine Inlet Pressure.	1	2	Q	SR 3.3.1.9[(b)(c)] SR 3.3.1.11[(b)(c)]	≤ 13% turbine power	Nominal 10% turbine power

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodology used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2.

(e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(f) Deleted

(g) Deleted

(h) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.2-1 (page 1 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1. Safety Injection						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.5	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2	NA	NA
c. Containment Pressure - High	1,2,3	3	E	SR 3.3.2.2 SR 3.3.2.6(b)(c)	≤ 4.5 psig	≤ 4.0 psig
d. Pressurizer Pressure - Low	1,2,3(a)	3	E	SR 3.3.2.1 SR 3.3.2.3(b)(c) SR 3.3.2.6(b)(c)	≥ 1725 psig	≥ 1730 psig
e. High Differential Pressure Between Steam Line Header and any Steam Generator (SG)	1,2,3(a)	3 per steam line	E	SR 3.3.2.1 SR 3.3.2.3(b)(c) SR 3.3.2.6(b)(c)	≤ 107 psi	≤ 100 psi

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodology used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2.

Table 3.3.2-1 (page 2 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1. Safety Injection (continued)						
f. Steam Line Flow - High	1,2,3$^{(d)}$	2 per steam line	E	SR 3.3.2.1 SR 3.3.2.3$^{(b)(c)}$ SR 3.3.2.6$^{(b)(c)}$	(e)	(f)
Coincident with T_{avg} - Low	1,2,3$^{(d)}$	1 per loop	E	SR 3.3.2.1 SR 3.3.2.3$^{(b)(c)}$ SR 3.3.2.6$^{(b)(c)}$	≥ 542.7°F	≥ 543.0°F
g. Steam Line Flow - High	1,2,3$^{(d)}$	2 per steam line	E	SR 3.3.2.1 SR 3.3.2.3$^{(b)(c)}$ SR 3.3.2.6$^{(b)(c)}$	(e)	(f)
Coincident with Steam Generator Pressure – Low	1,2,3$^{(d)}$	1 per SG	E	SR 3.3.2.1 SR 3.3.2.3$^{(b)(c)}$ SR 3.3.2.6$^{(b)(c)}$	≥ 607 psig$^{(i)}$	614 psig$^{(i)}$
2. Containment Spray						
a. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2	NA	NA

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodology used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2.

(d) Above the (T_{avg} - Low) interlock.

(e) Less than or equal to a function defined as ΔP corresponding to 41.2% steam flow at 0% load, and increasing linearly from 20% load to 114.4% steam flow at 100% load.

(f) Less than or equal to a function defined as ΔP corresponding to 40.0% steam flow at 0% load, and increasing linearly from 20% load to 114.0% steam flow at 100% load.

(i) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.

Table 3.3.2-1 (page 3 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
2. Containment Spray (continued)						
b. Containment Pressure - High High	1,2,3	3	E	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6(b)(c)	≤ 20.7 psig	≤ 20.0 psig
Coincident with Containment Pressure – High	1,2,3	3	E	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6(b)(c)	≤ 4.5 psig	≤ 4.0 psig
3. Containment Isolation						
a. Phase A Isolation						
(1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.5	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2	NA	NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
b. Phase B Isolation						
(1) Manual Initiation	1,2,3,4	2	F	SR 3.3.2.5	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2	NA	NA

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodology used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2.

Table 3.3.2-1 (page 4 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
3. Containment Isolation						
b. Phase B Isolation (continued)						
(3) Containment Pressure High High	1,2,3	3	I	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6[(b)(c)]	≤ 20.7 psig	≤ 20.0 psig
Coincident with Containment Pressure - High	1,2,3	3	I	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6[(b)(c)]	≤ 4.5 psig	≤ 4.0 psig
4. Steam Line Isolation						
a. Manual Initiation	1,2[(j)],3[(j)]	1 per steam line	J	SR 3.3.2.5	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2[(j)],3[(j)]	2 trains	D	SR 3.3.2.2	NA	NA
c. Containment Pressure - High - High	1,2[(j)],3[(j)]	3	I	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6[(b)(c)]	≤ 20.7 psig	≤ 20.0 psig
Coincident with Containment Pressure – High	1,2[(j)],3[(j)]	3	I	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6[(b)(c)]	≤ 4.5 psig	≤ 4.0 psig

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodology used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2

(j) Except when all MSIVs are closed and deactivated.

Table 3.3.2-1 (page 5 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
4. Steam Line Isolation (continued)						
d. Steam Line Flow – High	1,2[j],3[j]	2 per steam line	I	SR 3.3.2.1 SR 3.3.2.3[b][c] SR 3.3.2.6[b][c]	(e)	(f)
Coincident with T_{avg} – Low	1,2[j],3[j]	1 per loop	E	SR 3.3.2.1 SR 3.3.2.3[b][c] SR 3.3.2.6[b][c]	≥ 542.7°F	≥ 543.0°F
e. Steam Line Flow – High	1,2[j],3[j]	2 per steam line	I	SR 3.3.2.1 SR 3.3.2.3[b][c] SR 3.3.2.6[b][c]	(e)	(f)
Coincident with Steam Generator Pressure – Low	1,2[j],3[j]	1 per SG	I	SR 3.3.2.1 SR 3.3.2.3[b][c] SR 3.3.2.6[b][c]	≥ 607 psig[i]	614 psig[i]

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodology used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2.

(e) Less than or equal to a function defined as ΔP corresponding to 41.2% steam flow at 0% load, and increasing linearly from 20% load to 114.4% steam flow at 100% load.

(f) Less than or equal to a function defined as ΔP corresponding to 40.0% steam flow at 0% load, and increasing linearly from 20% load to 114.0% steam flow at 100% load.

(i) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.

(j) Except when all MSIVs are closed and deactivated.

Table 3.3.2-1 (page 6 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
5. Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	$1,2^{(k)},3^{(k)}$	2 trains	D	SR 3.3.2.2	NA	NA
b. SG Water Level – High High	$1,2^{(k)},3^{(k)}$	3 per SG	I	SR 3.3.2.1 SR 3.3.2.3$^{(b)(c)}$ SR 3.3.2.6$^{(b)(c)}$	≤ 80.5%	80.0%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	D	SR 3.3.2.2	NA	NA
b. SG Water Level - Low Low	1,2,3	3 per SG	E	SR 3.3.2.1 SR 3.3.2.3$^{(b)(c)}$ SR 3.3.2.6$^{(b)(c)}$	≥ 15.5%	16.0%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
d. Bus Stripping	1,2,3	1 per bus	G	SR 3.3.2.4 SR 3.3.2.6	NA	See LCO 3.3.5, "LOP EDG Start Instrumentation," for Trip Setpoints
e. Trip of all Main Feedwater Pumps Breakers	1,2	1 per breaker	H	SR 3.3.2.4	NA	NA

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodology used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2.

(k) Except when all MFIVs, MFRVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

Table 3.3.6-1 (page 1 of 2)
Containment Ventilation Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1.	Containment Isolation Phase A and Phase B					
	a. Manual Initiation	See LCO 3.3.2, "ESFAS Instrumentation," Table 3.3.2-1, Items 3.a.1 and 3.a.2, 3.a.3, 3.b.1, and 3.b.2 for containment ventilation isolation requirements.				
	b. Automatic Actuation Logic and Actuation Relays					
	c. Safety Injection					
2.	Containment Radiation					
	a. Gaseous	1[(a)],2[(a)] 3[(a)],4[(a)], (b)	1[(c)]	SR 3.3.6.1 SR 3.3.6.2[(d)(e)] SR 3.3.6.3[(d)(e)]	Refer to Note 1 (Page 3.3.6-5)	Refer to Note 1 (Page 3.3.6-5)
	b. Particulate	1[(a)],2[(a)] 3[(a)],4[(a)], (b)	1[(c)]	SR 3.3.6.1 SR 3.3.6.2[(d)(e)] SR 3.3.6.3[(d)(e)]	$\leq 9.45 \times 10^{-08}$ µCi/cc	$\leq 9.00 \times 10^{-08}$ µCi/cc

(a) Not applicable to containment purge supply and exhaust isolation valves.

(b) During movement of recently irradiated fuel assemblies within containment.

(c) Only one channel of either particulate or gaseous radioactivity channel is required.

(d) The instrument channel setpoint shall be reset to a value within the calibration tolerance of the Trip Setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable.

(e) If the instrument channel setpoint is less conservative than the Allowable Value, the setpoint shall be reset consistent with the Trip Setpoint and within 12 hours determine the affected channel is OPERABLE; otherwise, the channel shall be declared inoperable.

Table 3.3.6-1 (page 2 of 2)
Containment Ventilation Isolation Instrumentation

Note 1: Containment Radiation – Gaseous

Containment Gaseous Monitor Trip Setpoint = $\frac{(1.11 \times 10^{-3})}{(F)}$ µCi/cc,

Containment Gaseous Monitor Allowable Value = $\frac{(1.17 \times 10^{-3})}{(F)}$ µCi/cc,

Where $F = \frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in the Offsite Dose Calculation Manual.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.3	-------------------------------NOTE----------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.4	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.3.5	------------------------------NOTE-------------------------- Not required to be met for valves in penetrations isolated by blind flanges. --- Perform leakage rate testing for containment purge valves with resilient seals.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.3.6	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

4.0 DESIGN FEATURES

4.1 Site Location

The site is approximately 25 miles south of Miami, 8 miles east of Florida City and 9 miles southeast of Homestead, Florida.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 157 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy-4, **ZIRLO®**, **Optimized** ZIRLO™, or **AXIOM®** fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2), with or without dopants, as fuel material. Limited substitutions of stainless steel filler rods for fuel rods, or by vacant rod positions, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods.

4.2.2 Control Rod Assemblies

The reactor core shall contain 45 control rod assemblies. The control material shall be silver indium cadmium, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent,

b. $k_{eff} \leq 0.95$ if fully flooded with water borated to 550 ppm, which includes an allowance for biases and uncertainties as described in Section 9.5 of the UFSAR,

c. $k_{eff} \leq 1.0$ if fully flooded with unborated water, which includes an allowance for biases and uncertainties as described in Section 9.5 of the UFSAR,

d. A nominal 10.6 inch center to center distance between fuel assemblies placed in Region I of the fuel storage racks,

e. A nominal 9.0 inch center to center distance between fuel assemblies placed in Region II of the fuel storage racks, 5.5 Programs and Manuals

5.5.15 Control Room Envelope (CRE) Habitability Program (continued)

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation with one CREVS train operating at the flow rate required by the VFTP, at a Frequency in accordance with the Surveillance Frequency Control Program. The results shall be trended and used as part of the assessment of the CRE boundary.

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

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

5.5.16 Surveillance Frequency Control Program

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

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

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

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

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (continued)

1. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," S. L. Davidson and T. L. Ryan, April 1995.

2. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006.

3. WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016.

4. WCAP-18546-P-A, Revision 0, "Westinghouse AXIOM® Cladding for Use in Pressurized Water Reactor Fuel," March 2023.

The analytical methods used to determine Overtemperature ΔT and Overpower ΔT shall be those previously reviewed and approved by the NRC in:

1. WCAP-8745-P-A, "Design Basis for the Thermal Overtemperature ΔT and Overpower ΔT Trip Functions," September 1986.

2. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

The analytical methods used to determine Safety Limits, Shutdown Margin, Moderator Temperature Coefficient, DNB Parameters, Rod Bank Insertion Limits and the All Rods Out position shall be those previously reviewed and approved by the NRC in:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

The analytical methods used to support the suspension of the measurement of the Moderator Temperature Coefficient in accordance with Surveillance Requirement (SR) 3.1.3.2 shall be those previously reviewed and approved by the NRC in:

1. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997.

2. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988.

3. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (continued)

4. WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," August 2007

The ability to calculate the COLR nuclear design parameters are demonstrated in:

1. Florida Power & Light Company Topical Report NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants."

Topical Report NF-TR-95-01 was approved by the NRC for use by Florida Power & Light Company in:

1. Safety Evaluation by the Office of Nuclear Reactor Regulations Related to Amendment No. 174 to Facility Operating License DPR-31 and Amendment No. 168 to Facility Operating License DPR-41, Florida Power & Light Company Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.4 Post Accident Monitoring Report

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

5.6.5 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.6, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG;

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

5.6 Reporting Requirements

5.6.5 Steam Generator Tube Inspection Report (continued)

c. For each degradation mechanism found:

1. The nondestructive examination techniques utilized;

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

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

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

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

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

f. The results of any SG secondary side inspections;

g. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report;

h. The calculated accident induced leakage rate from the portion of the tubes below 18.11 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 1.82 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined; and

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 301 AND 294

TO SUBSEQUENT RENEWED FACILITY OPERATING LICENSE NOS. DPR-31 AND DPR-41

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING, UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

This safety evaluation (SE) contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations* (CFR) Section 2.390, "Public inspections, exemptions, requests for withholding." Proprietary information is identified by bold text enclosed within double brackets, as shown here: **[[example proprietary text]]**.

1.0 INTRODUCTION

By application dated November 15, 2023 (Reference 1), as supplemented by letters dated February 9, October 3, October 31, and November 12, 2024 (Reference 2, Reference 3, Reference 4, and Reference 5, respectively), Florida Power & Light Company (FPL, the licensee) requested changes to the technical specifications (TS) for Turkey Point Nuclear Generating Station (Turkey Point), Unit Nos. 3 and 4. The requested changes would revise the TS to increase certain surveillance requirement (SR) intervals from 18 months to 24 months for SRs whose frequency is controlled within the licensee's Surveillance Frequency Control Program (SFCP). The requested changes would also increase SR intervals for other TS within the Administrative Controls section of the TS. The licensee stated that all the requested changes are consistent with Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991 (Reference 7). In addition, the licensee requested a change that would revise the Surveillance Frequency specified in SR 3.6.3.5 from the SFCP to the Containment Leakage Rate Testing Program (CLRTP) of TS 5.5.13. The licensee also requested conforming changes to the Updated Final Safety Analysis Report (UFSAR).

By letter dated February 9, 2024, the licensee provided a supplement to its license amendment request (LAR), submitting technical report Westinghouse Commercial Atomic Program (WCAP)-18888-P, "Westinghouse Setpoint Methodology for Protection Systems, Turkey Point Units 3 & 4 24 Month Fuel Cycle," Revision 0 (Reference 8). WCAP-18888-P was prepared to combine two U.S. Nuclear Regulatory Commission (NRC, Commission)-approved setpoint methodologies into one comprehensive method for calculating TS-controlled instrument channel setpoints.

Enclosure 4

The supplemental letters dated February 9, October 3, October 31, and November 12, 2024, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 20, 2024 (89 FR 12873).

2.0 REGULATORY EVALUATION

2.1 Background

Improved reactor fuels allow licensees to consider an increase in the duration of the operating cycle for their facilities. The NRC staff has previously reviewed requests for individual plants to modify TS surveillance intervals to be compatible with a 24-month operating cycle. The NRC staff issued GL 91-04 to provide guidance to licensees for preparing such requests.

In its letter dated November 15, 2023, the licensee stated, in part:

> Florida Power and Light plans to extend selected Turkey Point Units 3 & 4 Surveillance Requirement (SR) intervals from the current 18-month to a maximum of 30-months (24-months plus 25 [percent] extension afforded by TS SR 3.0.2). Technical Specification (TS) SR changes are required to accommodate a proposed 24-month fuel cycle for Turkey Point Units 3 & 4. The proposed TS SR changes were evaluated in accordance with the guidance provided in Nuclear Regulatory Commission (NRC) Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. GL 91-04 provides the NRC Staff guidance that identifies the types of information that must be addressed when proposing extension of a SR intervals from 18 to 24-months.

In its letter dated November 15, 2023, the licensee provided the following system description:

> Turkey Point Updated Final Safety Analysis Report (UFSAR), Chapter 3, summarizes the current fuel design and application. Section 3.1.1 describes the performance objectives. The reactor core is currently a three-region cycled core. The fuel rods are cold worked partially annealed Zircaloy-4, ZIRLO® or Optimized ZIRLO™ tubes containing slightly enriched uranium dioxide fuel. All fuel rods are pressurized with helium during fabrication to reduce stresses and strains and to increase fatigue life.
>
> The fuel assembly consists of the rod cluster control (RCC) guide thimbles fastened to the grids and the top and bottom nozzles. The fuel rods are held in this assembly at seven points along their length by spring-clip grids which provide a very stiff support for the fuel rods.
>
> The Turkey Point Units are loaded with Westinghouse seven grid 15 Upgrade Assemblies (Upgrade) with the Westinghouse Integral Nozzle (WIN) as a replacement for the reconstitutable top nozzle (RTN). Full length rod cluster control assemblies (RCCA), secondary sources, thimble plug devices and burnable poison rods may be inserted into the guide thimbles of the fuel assemblies. The absorber sections of the control rods are fabricated of silver-

> indium-cadmium alloy sealed in stainless steel tubes. The absorber material in the fixed burnable poison rods is in the form of borosilicate glass sealed in stainless steel tubes.
>
> Three other types of burnable poison rods and absorbers are employed:
>
> a) Wet Annular Burnable Absorbers (WABA), each consisting of an aluminum oxide-boron carbide annulus sealed in Zircaloy,
>
> b) Reduced length Annular Hafnium Vessel Flux Depression (HVFD) absorbers which may be placed in peripheral assemblies as part of the flux reduction program, and
>
> c) Integral Fuel Burnable Absorbers (IFBA), consisting of a Zirconium diboride coating on the surface of the fuel pellets.

The licensee further stated the following in its letter dated November 15, 2023, regarding instrumentation and controls:

> Chapter 7 of the UFSAR summarizes the Instrument and Controls at Turkey Point. For the Reactor Protection System (RPS) description, typical core safety limits show the maximum trip points which are used for the protection system. The lines indicate a typical locus of the departure from nucleate boiling ratio (DNBR) equal to the safety analysis limit value at four pressures, and dashed lines indicate maximum permissible trip points for the RPS overtemperature-ΔT reactor trip. Actual setpoints are lower to allow for measurement and instrumentation errors. The RPS overpower-ΔT reactor trip limits the maximum core power independent of the DNBR.
>
> Adequate margins exist between the maximum nominal steady state operating point (which includes allowances for temperature, calorimetric, and pressure errors) and required trip points to preclude a spurious trip during design transients.

The licensee also stated that Section 7.2 of the UFSAR describes the current setpoint methodology. Specifically, the Trip Setpoints for the reactor trip system and engineering safety features are provided in UFSAR Table 7.2-1. The Trip Setpoint values are the Limiting Safety System Setting (LSSS) values that are calculated based on limits derived from the safety analyses and process instrumentation channel performance estimates and adjusted to account for the specific instrument channel uncertainties. The instrument uncertainties for the Trip Setpoints affected by the 2012 extended power uprate (EPU) had been based on the methodology described in WCAP-17070-P, “Westinghouse Setpoint Methodology for Protection Systems Turkey Point Units 3 and 4 (Power Uprate to 2644 MWt - Core Power).” The guidance of Technical Specifications Task Force (TSTF) No. 493, Revision 4, Option A, “Clarify Application of Setpoint Methodology for LSSS Functions,” was applied to the Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) Trip Setpoints and surveillance requirements impacted by the EPU.

The NRC staff understanding of the licensee's submittals is as follows:

The setpoint methodology establishes the Trip Setpoint and Allowable Value (AV) for each of the affected functions. The AVs at Turkey Point are "performance based" and are determined by adding (or subtracting) the rack calibration accuracy (RCA) of the instrument channel components tested during the Channel Operational Test (COT) to the Trip Setpoint in the non-conservative direction, i.e., toward, or closer to, the Safety Analysis Limit (SAL) for the application.

In accordance with the guidance of Generic Letter 91-04, for calibration interval extensions, a comparison of the projected drift errors over the new extended calibration interval was made by re-evaluating the values of instrument channel drift that should be used in the instrument channel setpoint uncertainty evaluations. Setpoint uncertainty evaluations conducted in support of the proposed TS changes due to the new maximum surveillance interval (including grace period) are identified in Section 3.1 below. No change to the safety analysis (i.e., analytical limit or other design basis assumption) is required to support the AV or Trip Setpoint changes.

The licensee chose to supplement its original submittal in February 2024, to more accurately reflect its updated instrument channel setpoint uncertainty methodology for TS-controlled instrument channel function setpoints. The new methodology is now referenced in Section 7.2 of the UFSAR as WCAP-18888-P, "Westinghouse Setpoint Methodology for Protection Systems Turkey Point Units 3 & 4 24 Month Fuel Cycle," Revision 0, dated January 2024, which combines the method for estimating channel uncertainty for all the TS-controlled instrument channel setpoints and AVs into one new setpoint methodology document. This enhances the clarity of the licensee's approach to establishing and maintaining its setpoints, since all TS-controlled setpoints will now be controlled in one document. (Previously, although not referenced in Section 7.2 of the UFSAR, the licensee had also used WCAP-12745-P, "Westinghouse Setpoint Methodology for Protection Systems – Turkey Point Units 3 & 4," (Reference 12) to establish those TS-controlled instrument channel function setpoints which were not impacted by the 2012 EPU and WCAP-17070-P (Reference 13) for instrument channel functions impacted by the 2012 EPU.)

2.2 Proposed Changes

2.2.1 Setpoint Methodology Update

In addition to proposing to update the surveillance frequencies to be lengthened with a typical surveillance extension from 18 to 24 months, the licensee also submitted a supplement to the license amendment in February 2024 that contained WCAP-18888-P. The purpose of WCAP-18888-P is to align and combine two previously NRC-approved setpoint methodologies into one technical document.

2.2.2 Increase SR Frequencies from 18 to 24 Months

The proposed amendment would modify the Turkey Point, Unit Nos. 3 and 4, TS to support a 24-month fuel cycle and would revise TS 5.5.16 for the SRs whose interval is contained in the licensee-controlled SFCP in accordance with Attachment 5, "Generic Letter 91-04 Evaluation," of Enclosure 1 to the LAR, which states, in part, that:

The SRs were broadly categorized as:

A. Non-calibration SRs

B. Calibration SRs without setpoints (no TS Allowable Values)

C. Calibration SRs with setpoints (TS Allowable Values)

Those SRs are described as follows:

<u>Non-Calibration SRs</u>

SR 3.1.4.2	SR 3.3.1.4	SR 3.3.1.5	SR 3.3.1.11	SR 3.3.2.2	SR 3.4.1.4
SR 3.4.9.2	SR 3.4.11.2	SR 3.4.14.2	SR 3.4.14.3	SR 3.5.2.7	SR 3.5.2.8
SR 3.6.6.3	SR 3.6.7.1	SR 3.6.7.2	SR 3.7.2.2	SR 3.7.5.3	SR 3.7.5.4
SR 3.8.1.8	SR 3.8.1.14	SR 3.8.1.15	SR 3.8.4.2	SR 3.8.4.3	

In addition to the TS surveillance frequencies controlled via the SFCP, there are three TS section 5.5 programs that currently have periodic 18-month frequency requirements that are not within the scope of the SFCP. The licensee additionally proposed to change these 18-month periodic requirements to 24 months. These three programs are:

- TS 5.5.2, “Primary Coolant Sources Outside Containment,”
- TS 5.5.8, “Ventilation Filter Testing Program (VFTP),” and

TS 5.5.2 includes integrated leak test requirements for systems within the scope of TS 5.5.2 “in accordance with the Surveillance Frequency Control Program.”

<u>Calibration SRs without Setpoints (with no associated TS Avs)</u>

SR 3.3.1.9	SR 3.3.2.6	SR 3.3.3.2	SR 3.3.4.3	SR 3.4.12.5	SR 3.4.15.3
SR 3.4.15.4					

<u>Calibration SRs with Setpoints (with associated TS Avs)</u>

SR 3.3.1.9	SR 3.3.1.10	SR 3.3.2.6	SR 3.3.6.3

The licensee stated that although there are no wording changes required for TS 5.5.2, the technical justification utilizing GL 91-04 was included in the LAR.

Additionally, TS 5.5.16, “Surveillance Frequency Control Program,” item b is proposed to be revised as follows (additions shown in bold underlined text):

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, “Risk-Informed Method for Control of Surveillance Frequencies,” Revision 1 **<u>or as specifically approved by the NRC</u>**.

2.2.3 Allowable Value and Trip Setpoint TS Changes

The licensee proposed the following TS changes:

TS table 3.3.1-1, "Reactor Trip System Instrumentation"

- Function 2. Power Range Neutron Flux, b. Low from "≤ 28%" rated thermal power (RTP) to "≤ 25.6%" RTP.
- Function 7. Pressurizer Pressure – Low from "≥ 1817" pounds per square inch gauge (psig) to "≥ 1830" psig.
- Function 8. Pressurizer Pressure – High from "≤ 2403" psig to "≤ 2390" psig.
- Function 14. Underfrequency RCPs [Reactor Coolant Pumps] Breakers Open from "≥ 55.9" Hertz (Hz) to "≥ 56.08" Hz

Footnotes (f) and (g) on TS pages 3.3.1-11, -12, -13, and -14 are proposed to be deleted in their entirety.

Footnote (c) on TS pages 3.3.1-11, -12, -13, and -14 is proposed to be revised as follows (additions shown in bold underlined text and deletions shown in double strikethrough text):

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the **methodology** ~~methodologies~~ used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2.

Additionally, TS table 3.3.1-1 functions that currently reference Footnotes (f) and (g), which are proposed to be deleted, are proposed to be changed to reference Footnotes (b) and (c).

TS table 3.3.2-1, "Engineered Safety Feature Actuation System Instrumentation"

- Function 1. Safety Injection, d. Pressurizer Pressure – Low from "≥ 1712" psig to "≥ 1725" psig.
- Function 1. Safety Injection, f. Coincident with T_{avg} [Reactor Coolant System average temperature] – Low from "≥ 542.5" degrees Fahrenheit (°F) to "≥ 542.7" °F.
- Function 2. Containment Spray, b. Containment Pressure – High High from "≤ 22.6" psig to "≤ 20.7" psig.
- Function 3. Containment Isolation, b. Phase B Isolation, (3) Containment Pressure High High from "≤ 22.6" psig to "≤ 20.7" psig.
- Function 4. Steam Line Isolation, c. Containment Pressure – High-High from "≤ 22.6" psig to "≤ 20.7" psig
- Function 4. Steam Line Isolation, d. Coincident with T_{avg} – Low from "≥ 542.5" °F to "≥ 542.7" °F.

TS table 3.3.2-1, pages 3.3.2-7 and 3.3.2-9, Footnotes (b) and (c) are proposed to be revised as follows (additions shown in bold underlined text and deletions shown in double strikethrough text):

(b) **<u>If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service</u>** ~~The instrument channel setpoint shall be reset to a value within the calibration tolerance of the Trip Setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable~~.

(c) **<u>The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodology used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2</u>** ~~If the instrument channel setpoint is less conservative than the Allowable Value, the setpoint shall be reset consistent with the Trip Setpoint and within 12 hours determine the affected channel is OPERABLE; otherwise, the channel shall be declared inoperable~~.

TS table 3.3.2-1, pages 3.3.2-8, 3.3.2-11, and 3.3.2-12, current Footnotes (b) and (c) are proposed to be deleted, Footnotes (g) and (h) are proposed to be renamed as Footnotes (b) and (c), and new Footnote (c) is proposed to be revised as follows (additions shown in bold underlined text and deletions shown in double strikethrough text):

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the **<u>methodology</u>** ~~methodologies~~ used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2.

Additionally, TS table 3.3.2-1 functions that currently reference Footnotes (g) and (h) are proposed to be changed to reference Footnotes (b) and (c), as Footnotes (g) and (h), respectively, were renamed.

TS table 3.3.2-1, page 3.3.2-10, Footnotes (b) and (c) are proposed to be revised as follows (additions shown in bold underlined text and deletions shown in double strikethrough text):

(b) **<u>If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service</u>** ~~The~~

~~instrument channel setpoint shall be reset to a value within the calibration tolerance of the Trip Setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable~~.

(c) **<u>The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodology used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2</u>** ~~If the instrument channel setpoint is less conservative than the Allowable Value, the setpoint shall be reset consistent with the Trip Setpoint and within 12 hours determine the affected channel is OPERABLE; otherwise, the channel shall be declared inoperable~~

<u>TS table 3.3.6-1, "Containment Ventilation Isolation Instrumentation"</u>

- Function 2. Containment Radiation, b. Particulate, Allowable Value from "≤ 5.00 X 10^{-6}" microcuries per cubic centimeter (µCi/cc) to "≤ 9.45 X 10^{-08}" µCi/cc.
- Function 2. Containment Radiation, b. Particulate, Trip Setpoint from "≤ 4.49 X 10^{-6}" µCi/cc to "≤ 9.00 X 10^{-08}" µCi/cc

Additionally, the licensee proposed to make the following change to "Note 1: Containment Radiation – Gaseous" (additions shown in bold underlined text and deletions in double strikethrough text):

Containment Gaseous Monitor Allowable Value = (**<u>1.17</u>**~~1.22~~ x 10^{-3})/(F) µCi/cc

2.2.4 <u>SR 3.6.3.5 Revision</u>

The proposed amendment would revise the surveillance frequency specified in SR 3.6.3.5 from being in accordance with the SFCP to being in accordance with the CLRTP of TS 5.5.13.

2.2.5 <u>TS 5.6.3, "CORE OPERATING LIMITS REPORT"</u>

The licensee proposed to delete the following core operating limit report (COLR) references from TS 5.6.3, section b:

1. WCAP-9220-P-A, Rev. 1, "Westinghouse ECCS Evaluation Model - 1981 Version," February 1982.

2. WCAP-10054-P-A, (proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

3. WCAP-10054-P-A, Addendum 2, Revision 1 (proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP

Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

4. WCAP-16009-P-A, "Realistic Large-break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.

5. USNRC Safety Evaluation Report, Letter from R. C. Jones (USNRC) to N. J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-12945(P) 'Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis,'" June 28, 1996 (as evaluated in NRC Safety Evaluation dated December 20, 1997).

6. Letter dated June 13, 1996, from N. J. Liparulo (W) to Frank R. Orr (USNRC), "Re-Analysis Work Plans Using Final Best Estimate Methodology," (as evaluated in NRC Safety Evaluation dated December 20, 1997).

The licensee also proposed to delete the following NOTES in TS 5.6.3, section b:

1. References 1 through 6 only applicable to Unit 3 through Core Operating Cycle 32 and Unit 4 through Core Operating Cycle 33.

2. Reference 9 not applicable to Unit 3 until Core Operating Cycle 33 and Unit 4 until Core Operating Cycle 34.

The licensee proposed to add the following reference to TS 5.6.3, section b as reference 4 and to renumber references 7, 8, and 9 as references 1, 2, and 3, respectively.

4. WCAP-18546-P-A, Revision 0, "Westinghouse AXIOM® Cladding for Use in Pressurized Water Reactor Fuel," March 2023.

2.2.6 TS 4.2, "Reactor Core"

The licensee proposed to revise TS 4.2.1, "Fuel Assemblies," as follows (additions shown in bold underlined text and deletions shown in double strikethrough text):

> The reactor shall contain 157 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy-4, ZIRLO®, ~~or~~ Optimized ZIRLO™**, or AXIOM®** fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2)**, with or without dopants**, as fuel material. Limited substitutions of stainless steel filler rods for fuel rods, or by vacant rod positions, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods.

Note: "ZIRLO®," "Optimized ZIRLO™," and "AXIOM®" are proposed to be shown as bold text.

2.3 Regulatory Requirements and Guidance

2.3.1 Regulatory Requirements

The regulations at Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.90, "Application for amendment of license, construction permit, or early site permit," state that whenever a holder of an operating license desires to amend the license, including TS in the license, an application for amendment must be filed with the Commission fully describing the changes desired. The regulations at 10 CFR 50.92(a) state that determinations on whether to grant an applied for license amendment are guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. These considerations include, as stated in 10 CFR 50.40(a), how the TS provide reasonable assurance that the health and safety of the public will not be endangered. Also, to issue an operating license, of which TS are a part, the Commission must make the findings of 10 CFR 50.57, "Issuance of operating license," including the 10 CFR 50.57(a)(3)(i) finding that there is reasonable assurance that the activities authorized by the operating license can be conducted without endangering the health and safety of the public.

The NRC staff also considered the following regulatory requirements related to this LAR:

The regulation at 10 CFR 50.36, "Technical specifications," which details the content and information that must be included in a facility's TS.

The regulation at 10 CFR 50.36(a)(1), which states, in part:

> Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section.

The regulation at 10 CFR 50.36(c)(1)(ii)(A), which states, in part:

> Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence.

The regulation at 10 CFR 50.36(c)(3), which states:

> Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The regulation at 10 CFR 50.36(c)(4), which states:

> Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of [10 CFR 50.36].

The regulation at 10 CFR 50.36(c)(5), which states, in part:

> 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.

The regulation at 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," which requires that preventative maintenance activities must not reduce the overall availability of the systems, structures, and components.

Key regulatory requirements specified in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," paragraph (a)(1) that are relevant to the LAR include the following:

- Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must perform analysis of core cooling performance under postulated loss-of-coolant accident (LOCA) conditions using an acceptable evaluation model (EM).

- An acceptable LOCA EM must be used that either applies realistic methods with an explicit accounting for uncertainties or follows the prescriptive, conservative requirements of Appendix K to 10 CFR Part 50.

- Core cooling performance must be analyzed for a number of postulated LOCAs of different sizes, locations, and other characteristics to ensure that the most severe event is calculated.

The emergency core cooling system (ECCS) acceptance criteria of 10 CFR 50.46(b)(1) though (b)(5) state, in part:

(1) *Peak cladding temperature*. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

(2) *Maximum cladding oxidation*. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) *Coolable geometry*. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) *Long-term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The General Design Criterion (GDC) used during the licensing of Turkey Point were based on the 1967 Atomic Energy Commission Proposed General Design Criterion (1967 Proposed GDC) (Reference 6) and predate 10 CFR Part 50, Appendix A. The following 1967 Proposed GDC were considered related to this LAR:

1967 Proposed GDC 6, "Reactor Core Design," which states:

> The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

1967 Proposed GDC 7, "Suppression of Power Oscillations," which states:

> The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed

1967 Proposed GDC 8, "Overall Power Coefficient," which states:

> The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

1967 Proposed GDC 10, "Containment," which states:

> Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

1967 Proposed GDC 12, "Instrumentation and Control Systems," which states:

> Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

1967 Proposed GDC 14, "Core Protection Systems," which states:

> Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

1967 Proposed GDC 15, "Engineered Safety Features Protection Systems," which states:

> Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

1967 Proposed GDC 19, "Protection Systems Reliability," which states:

> Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

1967 Proposed GDC 29, "Reactivity Shutdown Capability," which states:

> At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

1967 Proposed GDC 30, "Reactivity Holddown Capability," which states:

> At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

1967 Proposed GDC 31, "Reactivity Control Systems Malfunction," which states:

> The reactivity control systems shall be capable of sustaining any single malfunction, such as, unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

1967 Proposed GDC 32, "Maximum Reactivity Worth of Control Rods," which states:

> Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

1967 Proposed GDC 33, "Reactor Coolant Pressure Boundary Capability," which states:

> The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary

> component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

1967 Proposed GDC 34, "Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention," which states:

> The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

1967 Proposed GDC 38, "Reliability and Testability of Engineered Safety Features," which states:

> All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

1967 Proposed GDC 41, "Engineered Safety Features Performance Capability," which states:

> Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

1967 Proposed GDC 44, "Emergency Core Cooling System Capability," which states:

> At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or

> component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period of this function is required following the accident.

1967 Proposed GDC 49, "Containment Design Basis," which states:

> The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

The regulation at 10 CFR Part 50, Appendix J, states, in part, "primary reactor containments shall meet the containment leakage test requirements set forth in this appendix."

The regulation at 10 CFR Part 50, Appendix K, establishes required and acceptable features of EMs for heat removal by the ECCS after the blowdown phase of a LOCA. It consists of the following two parts:

- required and acceptable features of LOCA EMs, and
- documentation required for LOCA EMs.

The first part specifies modeling requirements and acceptable methods for simulating significant physical phenomena throughout all phases of a design-basis LOCA event, including relevant heat sources, fuel rod performance, and thermal-hydraulic (T-H) behavior. The second part specifies requirements for the documentation of LOCA EMs, including a complete description, a code listing, sensitivity studies, and comparisons against experimental data.

2.3.2 Regulatory Guidance

The NRC staff consulted the following guidance in its review of the LAR.

NRC NUREGs

NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [light-water reactor] Edition," including the sections given below:

- Section 4.3, "Nuclear Design," Revision 3, dated March 2007

- Section 4.4, "Thermal and Hydraulic Design," Revision 2, dated March 2007

- Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)," Revision 3, dated March 2007

- Section 6.2.2, "Containment Heat Removal Systems," Revision 5, dated March 2007

- Section 6.3, “Emergency Core Cooling System,” Revision 3, dated March 2007
- Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” Revision 4, dated March 2007
- Section 7.1-T, “Table 7-1 Regulatory Requirements, Acceptance Criteria, and Guidelines for Instrumentation and Control Systems Important to Safety,” Revision 6, dated August 2016
- Appendix 7.1-B, “Guidance for Evaluation of Conformance to IEEE Std 279,” Revision 6, dated August 2016
- Section 7.2, “Reactor Trip System,” Revision 6, dated August 2016
- Section 7.3, “Engineered Safety Features Systems,” Revision 6, dated August 2016
- Section 15.0, “Introduction – Transient and Accident Analyses,” Revision 3, dated March 2007

NRC Generic Communications

The NRC staff considered the regulatory guidance in GL 91-04, which discusses multiple criteria to be addressed when a licensee chooses to alter its refueling cycle from 18 to 24 months. GL 91-04, along with Enclosure 1, states that other instruments with an 18-month surveillance frequency requirement that are not instrument calibration related (i.e., non-calibration changes) should also be evaluated for the effect on safety associated with an extension to a 24-month required interval. This evaluation should address the following three criteria:

1. The licensee should analyze the effect on plant safety from the change in surveillance intervals to accommodate a 24-month fuel cycle. This evaluation should support a conclusion that the effect on safety is small.
2. The licensee should confirm that historical plant maintenance and surveillance data support the conclusion that the effect on safety is small.
3. The licensee should confirm that the performance of each surveillance at the bounding surveillance interval limit would not invalidate any assumption in the plant licensing basis.

For those SRs where the evaluation accomplishes these goals, the licensee need not quantify the effect of the change in surveillance intervals on the availability of individual systems or components. No change in the existence, testability, or availability of plant systems and components is being requested in the LAR, only an extension in the frequency of the tests or inspections.

GL 91-04, Enclosure 2, “Guidance for Addressing the Effect of Increased Surveillance Intervals on Instrument Drift and safety Analysis Assumptions,” identifies the following seven steps for the evaluation of instrumentation calibration related (i.e., calibration changes) frequency changes.

1. Confirm that instrument drift as determined by as-found and as-left calibration data from surveillance and maintenance records has not, except on rare occasions, exceeded acceptable limits for a calibration interval.

2. Confirm that the values of drift for each instrument type (make, model, and range) and application have been determined with a high probability and a high degree of confidence. Provide a summary of the methodology and assumptions used to determine the rate of instrument drift with time based upon historical plant calibration data.

3. Confirm that the magnitude of instrument drift has been determined with a high probability and a high degree of confidence for a bounding calibration interval of 30 months for each instrument type (make, model, number, and range) and application that performs a safety function. Provide a list of the channels by TS section that identifies these instrument applications.

4. Confirm that a comparison of the projected instrument drift errors has been made with the values of drift used in the setpoint analysis. If this results in revised setpoints to accommodate larger drift errors, provide proposed TS changes to update trip setpoints. If the drift errors result in a revised safety analysis to support existing setpoints, provide a summary of the updated analysis conclusions to confirm that safety limits and safety analysis assumptions are not exceeded.

5. Confirm that the projected instrument errors caused by drift are acceptable for control of plant parameters to effect a safe shutdown with the associated instrumentation.

6. Confirm that all conditions and assumptions of the setpoint and safety analyses have been checked and are appropriately reflected in the acceptance criteria of plant surveillance procedures for channel checks, channel functional tests, and channel calibrations.

7. Provide a summary description of the program for monitoring and assessing the effects of increased calibration surveillance intervals on instrument drift and its effect on safety.

<u>NRC Regulatory Guides (RG)</u>

RG 1.105, “Setpoints for Safety-Related Instrumentation,” Revision 4, dated February 2021 (Reference 9), describes a method acceptable to the NRC staff for complying with the NRC’s regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits. RG 1.105, Revision 4, endorses American National Standards Institute (ANSI)/International Society of Automation (ISA) Standard 67.04.01-2018, “Setpoints for Nuclear Safety-Related Instrumentation.” The NRC staff used this guidance to establish the adequacy of the licensee’s setpoint calculation methodology and the related plant surveillance procedures.

RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” dated March 2001. RG 1.190 was developed to provide state-of-the-art calculations and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence.

RG 1.236, “Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents,” dated June 2020. RG 1.236 describes methods and procedures that the NRC staff considers acceptable when analyzing the nuclear reactor’s initial response to a postulated control rod ejection (CRE) accident for pressurized-water reactors (PWRs).

Regulatory Issue Summaries (RIS)

RIS 2006-17, “NRC Staff Position on the Requirements of 10 CFR 50.36, ‘Technical specifications,’ Regarding Limiting Safety System Settings during Periodic Testing and Calibration of Instrument Channels” (Reference 10), addresses requirements on LSSSs that are assessed during the periodic testing and calibration of instrumentation. RIS 2006-17 discusses issues that could occur during the testing of LSSSs and that, therefore, may have an adverse effect on equipment operability. TSTF-493, Option A is one of many TSTFs for the Standard Technical Specifications for Westinghouse nuclear power plants where the licensee may choose to voluntarily adopt this TS change to address NRC concerns that the TS requirements for LSSSs may not be fully in compliance with the intent of 10 CFR 50.36 as described in RIS 2006-17.

3.0 TECHNICAL EVALUATION

3.1 Evaluation of TS-Controlled Setpoints and Allowable Values and Setpoint Methodology

The NRC staff reviewed the licensee’s proposed changes to various TS-controlled setpoints and AVs, as well as the licensee’s new setpoint methodology, WCAP-18888-P. A summary of the staff’s evaluation follows.

3.1.1 Proposed Changes to TS-Controlled Setpoints and Allowable Values

The licensee stated that it would use the revised AVs for specified functions in TS table 3.3.1-1, table 3.3.2-1, and table 3.3.6-1 consistent with the setpoint methodology for the trip setpoints affected by the 2012 EPU that was previously approved for Turkey Point, Unit Nos. 3 and 4. The setpoint methodology submitted by the licensee as part of the 2012 EPU, WCAP-17070-P, was evaluated and approved by the NRC as part of the EPU license amendment. The instant LAR proposed multiple changes to both TS-controlled setpoints and AVs.

The proposed changes to the AVs and the setpoint to change in support of a maximum surveillance interval of 30 months (including grace period) are listed below:

Proposed Changes to TS Allowable Values

- TS 3.3.1, Table 3.3.1-1, Function 2b Power Range Neutron Flux – Low Allowable Value would be revised from “≤ 28%” RTP to “≤ 25.6%” RTP.

- TS 3.3.1, Table 3.3.1-1, Function 7 Pressurizer Pressure – Low Allowable Value would be revised from “≥ 1817” psig to “≥ 1830” psig.

- TS 3.3.1, Table 3.3.1-1, Function 8 Pressurizer Pressure – High Allowable Value would be revised from “≤ 2403” psig to “≤ 2390” psig.

- TS 3.3.1, Table 3.3.1-1, Function 14 Underfrequency RCPs Breakers Open Allowable Value would be revised from "≥ 55.9" Hz to "≥ 56.08" Hz.

- TS 3.3.2, Table 3.3.2-1, Function 1d Safety Injection – Pressurizer Pressure – Low Allowable Value would be revised from "≥ 1712" psig to "≥ 1725" psig.

- TS 3.3.2, Table 3.3.2-1, Function 1f Safety Injection – Steam Line Flow – High Coincident with Tavg – Low Allowable Value would be revised from "≥ 542.5" °F to "≥ 542.7" °F.

- TS 3.3.2, Table 3.3.2-1, Function 2b Containment Spray – Containment Pressure - High High Allowable Value would be revised from "≤ 22.6" psig to "≤ 20.7" psig.

- TS 3.3.2, Table 3.3.2-1, Function 3b(3) Containment Isolation – Phase B Isolation – Containment Pressure High High Allowable Value would be revised from "≤ 22.6" psig to "≤ 20.7" psig.

- TS 3.3.2, Table 3.3.2-1, Function 4c Steam Line Isolation – Containment Pressure – High – High Allowable Value would be revised from "≤ 22.6" psig to "≤ 20.7" psig.

- TS 3.3.2, Table 3.3.2-1, Function 4d Steam Line Isolation – Steam Line Flow – High Coincident with Tavg – Low Allowable Value would be revised from "≥ 542.5" °F to "≥ 542.7" °F.

- TS 3.3.6, Table 3.3.6-1, Function 2a Containment Radiation – Gaseous Allowable Value would be revised from (1.22×10^{-3}) / F µCi/cc to (1.17×10^{-3}) / F µCi/cc.

- TS 3.3.6, Table 3.3.6-1, Function 2b Containment Radiation – Particulate Allowable Value would be revised from "≤ 5.00×10^{-6}" µCi/cc to "≤ 9.45×10^{-08}" µCi/cc.

<u>Proposed Change to TS Setpoints</u>

- TS 3.3.6, Table 3.3.6-1, Function 2b Containment Radiation – Particulate Trip Setpoint value would be revised from "≤ 4.49×10^{-6}" µCi/cc to "≤ 9.00×10^{-08}" µCi/cc.

As a result of these proposed changes, the NRC staff evaluated WCAP-18888-P and conducted a random sample of the affected TS Instrument Channel Functions to independently evaluate how the licensee's calculational methods were applied to determine the AVs and setpoints within its scope, as well as how the appropriate magnitude of the setpoint drift associated with each affected function was determined.

3.1.1 Setpoint Methodology Evaluation

WCAP-18888-P serves as an integration and modification technical report. It is an integration report in that it incorporates the information taken from two existing and previously NRC-approved setpoint methodologies for TS-controlled instrument channel function setpoints into one report. It is a modification report in that it brings the methodology described in WCAP-12745-P, "Westinghouse Setpoint Methodology for Protection Systems - Turkey Point Units 3 & 4" (Reference 12), originally developed in 1990 into alignment with the more modern setpoint methodology described in the second previously NRC-approved setpoint methodology WCAP-17070-P, “Westinghouse Setpoint Methodology for Protection Systems Turkey Point Units 3 & 4 (Power Uprate to 2644 MWt - Core Power),” Revision 1 (Reference 13), for a 2012 EPU that impacted eleven Turkey Point TS-controlled instrument channel setpoints. After the approval of the EPU license amendment in 2012, the remaining TS-controlled instrument channel setpoints continued to be controlled by the method described in WCAP-12745-P.

Concerning the establishment and maintenance of instrument channel setpoint and other setpoint related values within the scope of WCAP-12745-P, the licensee chose to update the instruments’ setpoint values to align with a setpoint establishment and maintenance process consistent with the process in WCAP-18888-P. In a general sense this application of the setpoint methodology resulted in a more conservative limiting value for those instruments previously controlled via the methodologies described in WCAP-12745-P. The setpoint methodology described in WCAP-17070-P is nearly identical to the setpoint methodology used in WCAP-18888-P and is consistent with the application of TSTF-493 Option A, “Clarify Application of Setpoint Methodology for LSSS Functions.”

In order to establish and maintain all TS-controlled setpoints using one consistent setpoint methodology as described in WCAP-18888-P via the adoption of a TSTF-493-like process (as direct reference to adoption of TSTF-493 Option A was not made in the LAR beyond its use for the setpoint methodology developed and implemented for the EPU (i.e., WCAP-17070-P)), in implementing WCAP-18888-P the licensee voluntarily chose to implement a setpoint establishment and maintenance process that is consistent with TSTF-493 Option A and the associated NRC RIS 2006-17, which provides the staff’s interpretation of how to properly implement 10 CFR 50.36 regarding LSSSs and instrument channel surveillances.

Specifically, the terms defined in WCAP-18888-P are consistent with those in TSTF-493 Option A and RIS 2006-17 and the approach used by TSTF-493, and the footnotes associated with the acceptance criteria used for ensuring LSSSs are appropriately maintained as addressed within TSTF-493, have been incorporated into the Turkey Point TS:

(a) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodology used to determine the as-found and as-left tolerances are specified in UFSAR Section 7.2.

In WCAP-18888-P, the licensee uses the square root sum of the squares (SRSS) method for combining the applicable random uncertainty terms, which it states is consistent with ANSI/ISA--67.04.01-2018. The licensee included the appropriate and applicable uncertainties for each RTS/ESFAS trip function in WCAP-18888-P. The total channel uncertainty (which the licensee defies as its channel statistical allowance (CSA)) represents a 95 percent probability, 95 percent confidence level (95/95) value as described in RG 1.105, Revision 4. In WCAP 18888-P, the licensee evaluated and confirmed that the RTS/ESFAS trip function uncertainty calculations are consistent with the guidance of RG 1.105, Revision 4, including using two-sided 95/95 tolerance limit acceptance criteria. In ANSI/ISA 67.04.01, the width of the random uncertainty tolerance interval (distance between the conservative and non-conservative direction 2-sigma tolerance interval endpoints) is determined by ensuring that total instrument channel random uncertainty encompasses at least 95 percent of the estimated channel uncertainty at a 95 percent confidence level. Bias uncertainties are algebraically combined before their addition to the estimate of random uncertainty to arrive at total instrument channel uncertainty.

Since the method utilized in WCAP-17070-P is consistent with that applied in WCAP-18888-P, additional information related to the specific terminology and application of that language for setpoint establishment and maintenance is located in the SE for the 2012 EPU in its "Instrument Setpoint Methodology" section (ML11293A365). These terms and phrases include as-found tolerance (AFT), as-left tolerance (ALT), nominal trip setpoint (NTS or NTSP) (and its relationship to the LSSS value), rack calibration accuracy (RCA), safety analysis limit (SAL), and total loop uncertainty (TLU). The NRC staff notes that some of the terms or phrases used in the Turkey Point setpoint methodologies (i.e., WCAP-17070-P and WCAP-18888-P) differ in their use when compared to RG 1.105, Revision 4, RIS 2006-17, and TSTF-493 Option A; however, the differences have been evaluated by the staff and determined to be adequate. For example, in WCAP-18888-P, the method chosen by the licensee results in the as-found tolerance equaling the as-left tolerance, which also equals the rack calibration accuracy, which are bi-directional tolerances that are symmetric about the established nominal trip setpoint, and whose magnitude also establishes the TS Allowable Value,[1] which is a one-sided operational limit applied in the non-conservative direction for the instrument channel. While establishing the magnitude of values for different terms (such as as-found tolerance and as-left tolerance) in the setpoint methodology using identical acceptance criteria values is not the approach outlined and applied in RIS 2006-17, or the industry standard endorsed in the RG, there is nothing stated in the RIS or the RG that would prohibit the appropriate use of such an approach.

The evaluation in Enclosure 4 to Attachment 1 of the LAR makes repeated references to WCAP-17070-P as a basis document for the evaluation. As stated earlier, in February 2024, the NRC staff received WCAP-18888-P for evaluation, which supersedes WCAP-17070-P. However, as the evaluation methodology used in WCAP-18888-P is nearly identical to that in WCAP-17070-P, the licensee's proposed evaluation methodology remains consistent and will be reviewed in Section 3.3 of this SE for setpoint uncertainty and drift analysis. Therefore, when conducting its evaluation of the LAR, the staff interpreted any reference to WCAP-17070-P as an evaluation against WCAP-18888-P, rather than WCAP-17070-P.

Based on its review, the NRC staff finds that the setpoint methodology described in WCAP-18888-P is acceptable for the LAR because application of the setpoint methodology for use in instrument channel uncertainty calculations will result in adequate margin between the AV and the analytical limit (AL), as well as adequate safety margin to accommodate an acceptable allowance for total instrument channel uncertainty between the NTS and AL.

[1] As defined in ANSI/ISA 67.04.01, the AV is established as the least conservative value of the "as-found" setpoint (i.e., the AFT) that a channel can have during a periodic TS-required channel calibration, channel operational test, or trip actuating device operational test requiring verification of the channel trip setpoint, beyond which appropriate action shall be taken as specified in the plant TS. Additionally, the AV is unidirectional in the non-conservative direction towards the SAL and not bi-directional as are the ALT and AFT.

3.1.3 Technical Conclusion

The method of developing performance-based setpoints and AVs in WCAP-18888-P, which combined the methods used in WCAP-17070-P and WCAP 12745-P, and which was first applied by using WCAP-17504, has been found acceptable by the NRC staff. Additionally, the application of those setpoints and AVs in the TS, along with the adoption of footnotes equivalent to those described in TSTF-493 for TS-controlled instrument channel setpoints, has been determined to be acceptable.

Therefore, the NRC staff concludes that the adoption of WCAP-18888-P and the incorporation of the updated TS provide reasonable assurance that functions associated with those setpoints and AVs will be initiated as required and in accordance with the applicable safety analyses and design bases.

3.2 Adoption of GL 91-04 Guidance

The NRC staff issued GL 91-04 to provide generic guidance to licensees in preparing amendment requests such as this LAR. The content of GL 91-04 must be adapted to the particular method by which the licensee chooses to categorize the specific surveillances within its TS. In the case of Turkey Point, the TS-controlled surveillances are broken down into three primary categories:

1) Non-calibration SRs,
2) calibration SRs without setpoints (no TS Allowable Values), and
3) calibration SRs with setpoints (TS Allowable Values).

3.2.1 Non-Calibration Changes

3.2.1.1 GL 91-04 Regulatory Guidance and Licensee Evaluation

GL 91-04 identifies three steps to evaluate non-calibration changes. The licensee provided the following general evaluations to those three steps.

STEP 1: Licensees should evaluate the effect on safety of an increase in 18-month surveillance intervals to accommodate a 24-month fuel cycle. This evaluation should support a conclusion that the effect on safety is small.

EVALUATION
Each proposed Non-Calibration SR interval change has been evaluated with respect to the effect on plant safety. The methodology utilized to justify the conclusion that changing the SR interval from an 18-month to a 24-month frequency has a minimal effect on safety, is based on whether the associated function/feature is:

1. Tested on a more frequent basis during the operating cycle by other plant programs;

2. Designed to have redundant counterparts or be single failure proof; or

3. Highly reliable.

A summary of the evaluation of the effect on safety for each proposed Non-Calibration SR interval change is presented [in the LAR].

STEP 2: Licensees should confirm that historical plant maintenance and surveillance data support this conclusion.

EVALUATION
The SR test history of the affected SRs has been evaluated. This evaluation consisted of a review of available SR test results and associated maintenance records going back 5 operating cycles. The evaluation documented in Attachment 2 [to the LAR] determined that current plant programs are adequate to ensure system reliability. SR failures that are discussed in Attachment 2 [to the LAR] exclude failures which:

1. Did not impact a TS safety function or TS operability;

2. Are detectable by required testing performed more frequently than the SR being extended; or

3. The cause can be attributed to an associated event such as a preventative maintenance task, human error, previous modification, or previously existing design deficiency; or that were subsequently re-performed successfully with no intervening corrective maintenance (e.g., plant conditions or malfunctioning measurement and test equipment may have caused aborting the test performance).

These types of failures are not related to potential unavailability due to SR interval extension and were therefore not further evaluated. This review of SR test history validates the conclusion that the impact, if any, on system availability will be minimal as a result of the change from an 18-month to a 24-month frequency. Specific SR test failures and justification for this conclusion are discussed in the [LAR].

STEP 3: Licensees should confirm that the assumptions in the plant licensing basis would not be invalidated as a result of performing any surveillance at the bounding surveillance interval limit provided to accommodate a 24-month fuel cycle.

EVALUATION
To confirm that the plant-licensing basis assumptions are not affected by the proposed increased surveillance interval limit from 18 to 24 months, a review of the Turkey Point Units 3 and 4 Updated Final Safety Analysis Report (UFSAR) and Regulatory Commitment data base was performed. This review confirmed that the proposed SR interval changes from 18 to 24 months do not impact the plant licensing basis.

3.2.2 Non-Calibration SR Interval Change Evaluation

The NRC staff reviewed the evaluations related to all non-calibration changes proposed by the licensee. The staff finds that the changes meet the guidance of GL 91-04 as explained under the following individual items and, therefore, are acceptable.

3.2.2.1 TS 3.1.4, "Rod Group Alignment Limits"

The LAR proposes to increase the interval of the following non-calibration SR from once every 18 months to once every 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2.

SR 3.1.4.2 Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.

A review of the SR history was conducted where no failures were found that would have impacted the TS functions that would be detected solely by the performance of this specific SR. Based on its review, the NRC staff finds that the effect on safety is small, and the SR extension is acceptable based on the guidance of GL 91-04.

3.2.2.2 TS 3.3.1, "Reactor Trip System (RTS) Instrumentation"

The LAR proposes to increase the interval of the following non-calibration SRs from once every 18 months to once every 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2.

SR 3.3.1.4 Perform TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) – Table 3.3.1-1 Function 19 – Reactor Trip Breakers (RTBs)

SR 3.3.1.5 Perform ACTUATION LOGIC TEST – Table 3.3.1-1 Function 20 – Automatic Trip Logic

SR 3.3.1.11 Perform CHANNEL OPERATIONAL TEST (COT) – Table 3.3.1-1 Function 17 – Reactor Trip System Interlocks

The NRC staff considered the three criteria associated with evaluating non-calibration SRs: (1) the effect on safety would be small, (2) historical data do not contradict this conclusion, and (3) no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

Table 3.3.1-1, Function 19, is associated with the Reactor Trip Breakers and the Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms.

Table 3.3.1-1, Function 20, is associated with the Automatic Trip and Interlock Logic for the reactor trip system using the automatic tester. The semiautomatic tester is used while the channel under test is in the bypass condition. The tester tests all possible logic combinations, with and without applicable permissives, while the interlock logic test verifies that the interlock is in its required state by observing the permissive annunciator window.

Table 3.3.1-1, Function 17, is associated with the conduct of a COT for multiple Reactor Trip System Interlocks. The test verifies that the channels operate as expected during plant startup and shutdown by simulating various signals at applicable power levels to ensure either a permissive or constraint (P&C) is enabled allowing the plant to continue to startup or shutdown as expected. This test occurs when the plant is undergoing a startup or shutdown, which will continue, but with a larger gap in time between testing due to the surveillance extension.

The licensee's statement that historical test records identify no failures of these TS functions (i.e., functional actuation of the reactor trip breakers and the related trip mechanisms), the fact that failures would have been detected exclusively by the conduct of these particular SRs, and the licensee's confirmatory review of the licensing basis in its UFSAR indicate that the impact on safety is small. Therefore, the NRC staff determined it is reasonable to conclude that extending the SR to a 24-month interval (+25 percent or 30 months maximum) is acceptable.

Based upon the NRC staff's review of the LAR, the changes meet the guidance of GL 91-04. The NRC staff finds the proposed interval extensions for the non-calibration SRs acceptable because: (1) the effect on safety would be small, (2) historical data do not contradict this conclusion, and (3) no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

3.2.2.3 TS 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation"

The LAR proposes to increase the interval of the following non-calibration SRs from once every 18 months to once every 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2.

SR 3.3.2.2 Perform ACTUATION LOGIC TEST – Table 3.3.2-1, Functions 1-6

Function 1 – Safety Injection
Function 2 – Containment Spray
Function 3 – Containment Isolation
Function 4 – Steam Line Isolation
Function 5 – Feedwater Isolation
Function 6 – Auxiliary Feedwater

For a more detailed description of these functions, refer to Table 3.3.2-1 of the Turkey Point TS.

In accordance with NUREG-1431, Vol. 1, Revision 5, "Standard Technical Specifications, Westinghouse Plants" (Reference 15), an Actuation Logic Test is the application of various simulated or actual input combinations in conjunction with each possible interlock logic state required for operability of a logic circuit and the verification of the required logic output. The Actuation Logic Test, as a minimum, shall include a continuity check of output devices.

The licensee reviewed the test history for these functions and determined that no failures of TS functions would have been identified exclusively by the performance of these types of SRs. It also performed a confirmatory review that verified that its licensing basis functions are not impacted by the SR interval extension, which caused it to conclude that the impact to safety is small. Based on its review, the NRC staff finds that the effect on safety is small, and the SR extension is acceptable based on the guidance of GL 91-04.

3.2.2.4 TS 3.4.1, “RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits”

The LAR proposes to increase the interval of the following non-calibration SR from once every 18 months to once every 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2.

SR 3.4.1.4 Verify by precision heat balance that RCS total flow rate is ≥ 270,000 gpm and greater than or equal to the limit specified in the COLR.

A review of the SR history was conducted where no failures were found that would have impacted the TS functions that would be detected solely by the performance of this specific SR. Based on its review, the NRC staff finds that the effect on safety is small, and the SR extension is acceptable based on the guidance of GL 91-04.

3.2.2.5 TS 3.4.9, “Pressurizer”

The LAR proposes to increase the interval of the following non-calibration SR from once every 18 months to once every 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2.

SR 3.4.9.2 Verify capacity of each required group of pressurizer heaters is ≥ 125 kW.

In the LAR, the licensee stated that the technical justification for the proposed change addresses the information contained in GL 91-04. For non-calibration SRs, GL 91-04 provides guidance to licensees on how to perform evaluations and confirmations as described in section 3.1.1 of this SE to support the change in surveillance intervals to accommodate a 24-month fuel cycle.

In the LAR, the licensee provided a summary of the evaluation to address the GL 91-04 guidance. Specifically, the licensee stated:

> This TS contains the requirements for maintaining required primary system pressure during steady state operation and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients. The pressure control components addressed by this LCO [limiting condition for operation] include the pressurizer water level, the required heaters, and their controls and emergency power supplies.
>
> SR 3.4.9.2 is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.

The licensee reviewed the SR test history and identified no failures of the TS functions that would have been solely detected by the periodic performance of this SR. The licensee stated at the impact, if any, on system availability is minimal from the proposed SR interval change from 18 to 24 months. The licensee determined that the impact of this change on safety, if any, is small.

The NRC staff reviewed the proposed change and determined that the proposed change for SR 3.4.9.2 is consistent with the intent of GL 91-04. The NRC staff finds the proposed interval extension for the above SR acceptable because: (1) the effect on safety would be small, (2) historical data do not contradict this conclusion, and (3) no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

3.2.2.6 TS 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)"

The LAR proposes to increase the interval of the following non-calibration SR from once every 18 months to once every 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2.

> SR 3.4.11.2 Perform a complete cycle of each PORV.

In the LAR, the licensee stated that the technical justification for the proposed change addresses the information contained in GL 91-04. For non-calibration SRs, GL 91-04 provides guidance to licensees on how to perform evaluations and confirmations as described in section 3.1.1 of this SE to support the change in surveillance intervals to accommodate a 24-month fuel cycle.

In the LAR, the licensee provided a summary of the evaluation to address the GL 91-04 guidance. Specifically, the licensee stated:

> This TS contains the requirements for maintaining automatic and operator-controlled pressurizer pressure relief capability by use of the PORVs. The PORVs are solenoid operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.
>
> SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for back-up pressure control during a SGTR [steam generator tube rupture].
>
> During performance of WO [work order] 4054277504 on 10/19/2018 PCV-3-456 was observed to fully open and fully close; however, the opening stroke time was 4.43 secs (1.65 to 3.05 acceptable) and closing time was 0.71 secs (1.0 to 2.0 acceptable). Stroke times were adjusted per procedure guidance and retest performed to satisfy IST [inservice test] requirements with satisfactory results.
>
> Unit 3 procedures dating from April 2017 through April 2023 were reviewed during performance of the SFA [surveillance failure analysis] for SR 3.4.11.2 with no additional issues noted. No issues noted with unit 4 procedures.

The licensee reviewed the SR test history and identified no failures of the TS functions that would have been solely detected by the periodic performance of this SR. The licensee stated that the impact, if any, on system availability is minimal from the proposed SR interval change from 18 to 24 months. The licensee determined the impact of this change on safety, if any, is small.

The NRC staff reviewed the proposed change and determined that the proposed change for SR 3.4.11.2 is consistent with the intent of GL 91-04. The NRC staff finds the proposed interval extensions for the above SR acceptable because: (1) the effect on safety would be small, (2) historical data do not contradict this conclusion, and (3) no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

3.2.2.7 TS 3.4.14, “RCS Pressure Isolation Valve (PIV) Leakage”

The LAR proposes to increase the interval of the following non-calibration SRs from once every 18 months to once every 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2.

> SR 3.4.14.2 Verify RHR [Residual Heat Removal] System autoclosure interlock prevents the valves from being opened with a simulated or actual RCS pressure signal ≥ 525 psig.
>
> SR 3.4.14.3 Verify RHR System autoclosure interlock causes the valves to close automatically with a simulated or actual RCS pressure signal ≥ 525 psig.

In the LAR, the licensee stated that the technical justification for the proposed change addresses the information contained in GL 91-04. For non-calibration SRs, GL 91-04 provides guidance to licensees on how to perform evaluations and confirmations as described in section 3.1.1 of this SE to support the change in surveillance intervals to accommodate a 24-month fuel cycle.

In the LAR, the licensee provided a summary of the evaluation to address the GL 91-04 guidance. Specifically, the licensee stated:

> General Design Criteria (GDC) 53, defines RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high-pressure RCS from an attached low-pressure system.
>
> PIVs are provided to isolate the RCS from the following typically connected systems:
>
> (a) Residual Heat Removal (RHR) System,
> (b) Safety Injection System, and
> (c) Chemical and Volume Control System.
>
> SR 3.4.14.2 verifies that the RHR System autoclosure interlocks are OPERABLE to ensure that RCS pressure will not pressurize the RHR system beyond 125 [percent] of its design pressure of 600 psig. The interlock setpoint is set so the

> actual RCS pressure must be < 525 psig to open the RHR [i]solation valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift.
>
> …
>
> SR 3.4.14.3 verifies the RHR System autoclosure interlocks are OPERABLE to ensure that RCS pressure will not pressurize the RHR system beyond 125 [percent] of its design pressure of 600 psig. The interlock setpoint is set so the RHR [i]solation valves will automatically close prior to actual RCS pressure exceeding 525 psig. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift.

The licensee reviewed the SR test history and identified no failures of the TS functions that would have been solely detected by the periodic performance of these SRs. The licensee stated that the impact, if any, on system availability is minimal from the proposed SR interval change from 18 to 24 months. The licensee determined that the impact of this change on safety, if any, is small.

The NRC staff reviewed the proposed change and determined that the proposed change for SRs 3.4.14.2 and 3.4.14.3 is consistent with the intent of GL 91-04. The NRC staff finds the proposed interval extensions for the above SRs acceptable because: (1) the effect on safety would be small, (2) historical data do not contradict this conclusion, and (3) no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

3.2.2.8 TS 3.5.2, “ECCS – Operating”

The LAR proposes to increase the interval of the following non-calibration SRs from once every 18 months to once every 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2.

> SR 3.5.2.7 Verify, for each ECCS throttle valve listed below, each position stop is in the correct position.
>
> Valve Number
> HCV-*-758
> MOV-*-872

The licensee stated that realignment of valves in the flow path on a safety injection (SI) signal is necessary for proper ECCS performance and that the valves associated with SR 3.5.2.7 have stops to allow proper positioning for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow.

A review of the SR history was conducted where no failures were found that would have impacted the TS functions that would be detected solely by the performance of this specific SR. Based on its review, the NRC staff finds that the effect on safety is small, and the SR extension is acceptable based on the guidance of GL 91-04.

SR 3.5.2.8 Verify, by visual inspection, each ECCS containment sump suction inlet is not restricted by debris and the suction components show no evidence of structural distress or abnormal corrosion.

The licensee stated that SR 3.5.2.8 requires performance of periodic inspections to verify that the containment sump does not show current or potential debris blockage, structural damage, or abnormal corrosion. This is to ensure the operability and structural integrity of the containment sump. The licensee stated that the strainer modules are functionally equivalent to trash racks and screens.

A review of the SR history was conducted where no failures were found that would have impacted the TS functions that would be detected solely by the performance of this specific SR. Based on its review, the NRC staff finds that the effect on safety is small, and the SR extension is acceptable based on the guidance of GL 91-04.

3.2.2.9 TS 3.6.6, "Containment Spray and Cooling Systems"

The LAR proposes to increase the interval of the following non-calibration SR from once every 18 months to once every 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2.

SR 3.6.6.3 Verify each emergency containment cooling unit cooling water flow rate is ≥ 2000 gpm.

In the LAR, the licensee stated that the technical justification for the proposed change addresses the information contained in GL 91-04. For non-calibration SRs, GL 91-04 provides guidance to licensees on how to perform evaluations and confirmations as described in section 3.1.1 of this SE to support the change in surveillance intervals to accommodate a 24-month fuel cycle.

In the LAR, the licensee provided a summary of the evaluation to address the GL 91-04 guidance. Specifically, the licensee stated:

> The Containment Spray and Emergency Containment Cooling systems provide containment atmosphere cooling to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray System and Containment Cooling System, were designed to meet Criterion 52, "Containment Heat Removal Systems," Criterion 58, "Inspection of Containment Pressure-Reducing Systems," Criterion 59, "Testing of Containment Pressure-Reducing Systems Components," Criterion 60, "Testing of Containment Spray Systems," Criterion 61, "Testing of Operational Sequence of Containment Pressure-Reducing Systems," and Criterion 62, "Inspection of Air Cleanup Systems," or other documents that were appropriate at the time of licensing (identified on a unit specific basis).

> The Emergency Containment Cooling System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post-accident period can be attained. The Containment Spray System and the Emergency Containment Cooling System provide a method to limit and maintain post-accident conditions to less than the containment design values.
>
> Three units of emergency containment cooling are provided. Each fan unit is supplied with cooling water from a train of component cooling water (CCW). Air is drawn into the coolers through the fan and discharged to the upper areas of containment.
>
> In post-accident operation following an actuation signal, two Emergency Containment Cooling System fans are designed to start automatically if not already running. Only two of the three emergency containment fans start on a safety injection signal (Containment High-1 pressure setpoint). The third emergency containment fan is required to be available and will automatically start if there is a failure of one of the other units. The temperature of the CCW is an important factor in the heat removal capability of the fan units.
>
> SR 3.6.6.3 verifies that CCW cooling flow rate to each emergency containment cooling unit is ≥2000 gpm to provide assurance that the design flow rate assumed in the safety analyses will be achieved.

The licensee reviewed the SR test history and identified no failures of the TS functions that would have been solely detected by the periodic performance of this SR. The licensee stated that the impact, if any, on system availability is minimal from the proposed SR interval change from 18 to 24 months. The licensee determined that the impact of this change on safety, if any, is small.

The NRC staff reviewed the proposed change and determined that the proposed change for SR 3.6.6.3 is consistent with the intent of GL 91-04. The NRC staff finds the proposed interval extensions for the above SR acceptable because: (1) the effect on safety would be small, (2) historical data do not contradict this conclusion, and (3) no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

3.2.2.10 TS 3.6.7, "Recirculation pH Control System"

The LAR proposes to increase the interval of the following non-calibration SRs from once every 18 months to once every 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2.

SR 3.6.7.1	Verify the buffering agent baskets are in place and intact.
SR 3.6.7.2	Verify the buffering agent baskets collectively contain > 7500 pound (154 cubic feet) of sodium tetraborate decahydrate, or equivalent.

In the LAR, the licensee stated that the technical justification for the proposed change addresses the information contained in GL 91-04. For non-calibration SRs, GL 91-04 provides guidance to licensees on how to perform evaluations and confirmations as described in section 3.1.1 of this SE to support the change in surveillance intervals to accommodate a 24-month fuel cycle.

In the LAR, the licensee provided a summary of the evaluation to address the GL 91-04 guidance. Specifically, the licensee stated:

> The Recirculation pH Control System is a passive safeguard consisting of 10 stainless steel wire mesh baskets (2 large and 8 small) containing sodium tetra borate decahydrate (NaTB) located in the containment basement (14' elevation). The initial containment spray will be boric acid solution from the refueling water storage tank. The Recirculation pH Control System adds NaTB to the containment sump when the level of boric acid solution from the containment spray and the coolant lost from the Reactor Coolant System rises above the bottom of the buffering agent baskets. As the sump level rises, the NaTB will begin to dissolve. The addition of NaTB from the buffering agent baskets ensures the containment sump pH will be greater than 7.0. The resultant alkaline pH of the spray enhances the ability of the recirculated spray to scavenge fission products from the containment atmosphere. The alkaline pH in the recirculation sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on stainless steel piping systems exposed to the solution.
>
> SR 3.6.7.1 verifies the buffering agent baskets are in place and intact to provide assurance that the system is able to provide additive to the containment sump in the event of a DBA. This verification ensures the NaTB baskets are in the proper location, the leveling feet are in the proper position, the baskets are at the proper height off the floor, and basket covers are installed.
>
> …
>
> To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the amount of NaTB must be sufficient to adjust pH for all recirculated water. SR 3.6.7.2 is performed to verify the amount of NaTB.

The licensee reviewed the SR test history and identified no failures of the TS functions that would have been solely detected by the periodic performance of these SRs. The licensee stated that the impact, if any, on system availability is minimal from the proposed SR interval change from 18 to 24 months. The licensee determined the impact of this change on safety, if any, is small.

The NRC staff reviewed the proposed change and determined that the proposed change for SRs 3.6.7.1 and 3.6.7.2 is consistent with the intent of GL 91-04. The NRC staff finds the proposed interval extensions for the above SRs acceptable because: (1) the effect on safety would be small, (2) historical data do not contradict this conclusion, and (3) no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

3.2.2.11 TS 3.7.2, "Main Steam Isolation Valves (MSIVs)"

The LAR proposes to increase the interval of the following non-calibration SR from once every 18 months to once every 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2.

> SR 3.7.2.2 Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.

In the LAR, the licensee stated that the technical justification for the proposed change addresses the information contained in GL 91-04. For non-calibration SRs, GL 91-04 provides guidance to licensees on how to perform evaluations and confirmations as described in section 3.1.1 of this SE to support the change in surveillance intervals to accommodate a 24-month fuel cycle.

In the LAR, the licensee provided a summary of the evaluation to address the GL 91-04 guidance. Specifically, the licensee stated:

> The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.
>
> The MSIVs close on a main steam isolation signal generated by 1) Steam Line Flow – High coincident with either Steam Generator Pressure – Low or Tavg – Low, or 2) Containment Pressure – High High coincident with Containment Pressure – High. The MSIVs fail closed on loss of control or actuation power.
>
> Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.
>
> SR 3.7.2.2 verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage.

The licensee reviewed the SR test history and identified no failures of the TS functions that would have been solely detected by the periodic performance of this SR. The licensee stated that the impact, if any, on system availability is minimal from the proposed SR interval change from 18 to 24 months. The licensee determined that the impact of this change on safety, if any, is small.

The NRC staff reviewed the proposed change and determined that the proposed change for SR 3.7.2.2 is consistent with the intent of GL 91-04. The NRC staff finds the proposed interval extensions for the above SR acceptable because: (1) the effect on safety would be small, (2) historical data do not contradict this conclusion, and (3) no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

3.2.2.12 TS 3.7.5, “Auxiliary Feedwater (AFW) System”

The LAR proposes to increase the interval of the following non-calibration SRs from once every 18 months to once every 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2.

SR 3.7.5.3 Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

SR 3.7.5.4 Verify each AFW pump starts automatically on an actual or simulated actuation signal.

In the LAR, the licensee stated that the technical justification for the proposed change addresses the information requested by GL 91-04. For non-calibration SRs, GL 91-04 provides guidance to licensees on how to perform evaluations and confirmations as described in section 3.1.1 of this SE to support the change in surveillance intervals to accommodate a 24-month fuel cycle.

In the LAR, the licensee provided a summary of the evaluation to address the GL 91-04 guidance. Specifically, the licensee stated:

> The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through suction lines from the condensate storage tank (CST) (LCO 3.7.6, “Condensate Storage Tank (CST)”) and pump to the steam generator secondary side via separate and independent connections to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1, “Main Steam Safety Valves (MSSVs)”) or atmospheric dump valves. If the main condenser is available, steam may be released via the steam bypass valves and recirculated to the CST.
>
> Upon loss of normal feedwater, steam is supplied to the AFW System from the unit which has lost feedwater. Steam can also be supplied from the opposite unit steam generators or from the unit’s auxiliary steam system. The unit supply valves will automatically open by any one of the following:
>
> a. Safety injection,
> b. Low-low level in any of the three steam generators,
> c. Loss of both feedwater pumps under normal operating conditions,
> d. Loss of AC [alternating current] electrical distribution 4.16 kV or 480 load center buses, and
> e. Anticipated Transient Without Scram (ATWS) Mitigating System Actuation Circuity (AMSAC) signal.
>
> The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit

to RHR entry conditions, with steam released through the ADVs [atmospheric dump valves].

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

SR 3.7.5.3 verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

…

The AFW System is a shared system between Units 3 and 4. The AFW System consists of three steam turbine driven pumps configured into two trains. Each pump provides 100 [percent] of required flow capacity to the steam generators, as assumed in the accident safety analysis. The three pumps are configured such that each can supply auxiliary feedwater to either Unit 3 or 4, with any single pump supplying the total feedwater requirement to both units. Steam supply to the three pumps is supplied from each unit steam generator via redundant steam supply headers. Each pump turbine can be manually aligned to either steam supply header. The steam supply line from each steam generator to both steam headers consists of a check valve and motor operated steam supply valve. The three pumps discharge through check valves to one of two redundant discharge headers. The AFW System is normally configured with one turbine drive pump aligned to Train 1 steam and feedwater headers and two turbine drive pumps aligned to Train 2 steam and feedwater headers. Auxiliary feedwater can be supplied through redundant lines to the safety related portions of the main feedwater lines to each of the steam generators. Each pump has sufficient capacity for single and two unit operation to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal (RHR) System may be placed into operation.

SR 3.7.5.4 verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal.

The licensee reviewed the SR test history and identified no failures of the TS functions that would have been solely detected by the periodic performance of this SR. The licensee stated that the impact, if any, on system availability is minimal from the proposed SR interval change from 18 to 24 months. The licensee determined that the impact of this change on safety, if any, is small.

The NRC staff reviewed the proposed change and determined that the proposed change for SRs 3.7.5.3 and 3.7.5.4 is consistent with the intent of GL 91-04. The NRC staff finds the proposed interval extensions for the above SRs acceptable because: (1) the effect on safety would be small, (2) historical data do not contradict this conclusion, and (3) no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

3.2.2.13 TS 3.8.1, "AC Sources – Operating"

The LAR states, in part, that the electrical power distribution system AC sources consist of the offsite power sources (startup transformers) and the onsite standby power sources (emergency diesel generators (EDGs)). Offsite circuits and EDGs are shared between the units. The normal power source to the Class 1E electrical power distribution system is the respective unit auxiliary transformers and associated circuits to the Train A and Train B 4.16 kilo Volt (kV) buses. The onsite standby power source for each 4.16 kV bus is a dedicated EDG. Two EDGs provide onsite emergency AC power for each unit.

The SR interval of the following non-calibration SR is proposed to be changed from 18 to 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2.

SR 3.8.1.8 Verify manual transfer of AC power sources from the auxiliary transformer to the startup transformer.

SR 3.8.1.14 Verify each EDG operates for ≥ 24 hours:

a. For ≥ 2 hours loaded ≥ 2550 kW and ≤ 2750 kW (Unit 3), ≥ 2950 kW and ≤ 3150 kW (Unit 4) and

b. For the remaining hours of the test loaded ≥ 2300 kW and ≤ 2500 kW (Unit 3), ≥ 2650 kW and ≤ 2850 kW (Unit 4).

SR 3.8.1.15 Verify each EDG starts and achieves:

a. In ≤ 15 seconds, voltage ≥ 3950 V and frequency ≥ 59.4 Hz and

b. Steady state voltage ≥ 3950 V, and ≤ 4350 V and frequency ≥ 59.4 Hz and ≤ 60.6 Hz.

In the LAR, the licensee stated that the proposed changes were evaluated in accordance with the guidance contained in GL 91-04, which provides guidance for identifying the types of information to be addressed when proposing the extension of SR intervals from 18 to 24 months.

The NRC staff notes that for non-calibration SRs, GL 91-04 provides that licensees should perform the following to support surveillance intervals to accommodate a 24-month fuel cycle:

a) Evaluate the effect on safety of the change in surveillance intervals to support a conclusion that the effect on safety is small.

b) Confirm that historical maintenance and surveillance data do not invalidate the conclusion that the effect on safety is small.

c) Confirm that the performance of surveillances at the bounding surveillance interval limit provided to accommodate a 24-month fuel cycle would not invalidate any assumption in the plant licensing basis.

For each corresponding provision of GL 91-04 above, in the LAR, the licensee provided, in part, the evaluation as followed:

a) Each proposed non-calibration SR interval change has been evaluated with respect to the effect on plant safety. The methodology utilized to justify the conclusion that changing the SR interval from an 18-month to a 24-month frequency has a minimal effect on safety, is based on whether the associated function/feature is:

- Tested on a more frequent basis during the operating cycle by other plant programs;
- Designed to have redundant counterparts or be single failure proof; or
- Highly reliable.

b) Historical review of SR test results and associated maintenance records did not find evidence of failures that would invalidate the conclusion that the effect on safety is small.

c) A review of the Turkey Point licensing basis confirmed that plant-licensing basis assumptions are not affected by the proposed SR interval changes.

By letter dated October 3, 2024 (Reference 3), in response to an NRC staff request for additional information (RAI), the licensee provided the following SRs that are tested on a more frequent basis to further support the conclusion that the effect of the proposed change on safety is small:

SR 3.8.1.1 Verify correct breaker alignment and indicated power availability for each offsite circuit (every 31 days).

SR 3.8.1.2 Verify each EDG starts from standby conditions and achieves steady state voltage ≥ 3950 V and ≤ 4350 V, and frequency ≥ 59.4 Hz and ≤ 60.6 Hz (every 31 days).

SR 3.8.1.3 Verify each EDG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2300 kW and ≤ 2500 kW (Unit 3), ≥ 2650 kW and ≤ 2850 kW (Unit 4) (every 31 days).

SR 3.8.1.7 Verify each EDG starts from standby condition and achieves in ≤ 15 seconds, voltage ≥ 3950 V and ≤ 4350 V, and frequency ≥ 59.4 Hz and ≤ 60.6 Hz (every 184 days).

The NRC staff notes that the above more frequently tested SRs could maintain the ability of the AC power source supporting safety functions necessary to safely shut down the reactor and maintain the reactor in a safe shutdown condition. Therefore, the NRC staff determined that the licensee's evaluation addressing GL 91-04 provides reasonable assurance that the effect on safety of the change in surveillance intervals is small.

The NRC staff reviewed the LAR, as supplemented, the UFSAR, and the TS bases and determined that the proposed change for the above SRs is consistent with the intent of GL 91-04. The NRC staff finds the proposed interval extensions for SR 3.8.1.8, SR 3.8.1.14, and SR 3.8.1.15 acceptable because the effect on safety would be small, historical data does not contradict this conclusion, and no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

3.2.2.14 TS 3.8.4, "DC Sources – Operating"

In the LAR, the licensee stated, in part, that the direct current (DC) electrical power sources are shared between units. The DC electrical power system contains five safety-related 125 V batteries and associated battery chargers. Two battery banks are associated with each unit, one 1800 ampere-hour (AH) and one 1200 AH, and a spare 1945 AH battery bank that can be substituted, to allow for testing or maintenance, for any of the other four battery banks. Each 1800 AH battery bank has two safety-related full capacity 400 ampere solid-state battery chargers associated with it (one normal charger and one alternate charger). Each 1200 AH battery bank has two safety-related full capacity 300 ampere solid-state battery chargers associated with it (one normal charger and one alternate charger). The spare battery bank is normally isolated from the vital DC buses and maintained in a fully charged condition by a non-safety related battery charger. Each battery has been sized to support operation of its required loads for 2 hours without terminal voltage falling below its minimum required value. The SR interval of the following non-calibration SRs is proposed to be changed from 18 to 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2.

SR 3.8.4.2 Verify each battery charger supplies ≥ 400 amps (battery chargers associated with battery banks 3A and 4B) and ≥ 300 amps (battery chargers associated with battery banks 3B and 4A) at greater than or equal to the minimum established float voltage for ≥ 8 hours.

<u>OR</u>

Verify each battery charger can recharge the battery to the fully charged state within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.

SR 3.8.4.3 Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.

For non-calibration SRs, GL 91-04 provides that licensees should perform the following to support surveillance intervals to accommodate a 24-month fuel cycle:

a) Evaluate the effect on safety of the change in surveillance intervals to support a conclusion that the effect on safety is small.

b) Confirm that historical maintenance and surveillance data do not invalidate the conclusion that the effect on safety is small.

c) Confirm that the performance of surveillances at the bounding surveillance interval limit provided to accommodate a 24-month fuel cycle would not invalidate any assumption in the plant licensing basis.

For each corresponding provision of GL 91-04 above, in the LAR, the licensee provided, in part, the evaluation as followed:

a) Each proposed non-calibration SR interval change has been evaluated with respect to the effect on plant safety. The methodology utilized to justify the conclusion that changing the SR interval from an 18-month to a 24-month frequency has a minimal effect on safety, is based on whether the associated function/feature is:

- Tested on a more frequent basis during the operating cycle by other plant programs;
- Designed to have redundant counterparts or be single failure proof; or
- Highly reliable.

b) Historical review of SR test results and associated maintenance records did not find evidence of failures that would invalidate the conclusion that the effect on safety is small.

c) A review of the Turkey Point licensing basis confirmed that plant-licensing basis assumptions are not affected by the proposed SR interval changes.

By letter dated October 3, 2024 (Reference 3), in response to an NRC staff RAI, the licensee provided the following SR and Technical Requirement Surveillance (TRS) that are tested on a more frequent basis to further support the conclusion that the effect of the proposed change on safety is small:

SR 3.8.4.1 Verify battery terminal voltage is greater than or equal to the minimum established float voltage (every 31 days).

TRS 13.8.3 Verify each battery charger is supplying ≥ 10 amperes to the associated battery bank (every 24 hours).

The NRC staff notes that the above more frequently tested SR and TRS could maintain the ability of the DC power source supporting safety functions necessary to safely shutdown the reactor and maintain the reactor in a safe shutdown condition. Therefore, the NRC staff determined that the licensee's evaluation addressing GL 91-04 provides reasonable assurance that the effect on safety of the change in surveillance intervals is small.

The NRC staff reviewed the LAR, as supplemented, the UFSAR, and the TS bases and determined that the proposed change for the above SRs is consistent with the intent of GL 91-04. The NRC staff finds the proposed interval extensions for SR 3.8.4.2 and SR 3.8.4.3 acceptable because the effect on safety would be small, historical data does not contradict this conclusion, and no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

3.2.2.15 TS 5.5.2, “Primary Coolant Sources Outside Containment”

TS 5.5.2 states, in part, that the Primary Coolant Sources Outside Containment program shall include the following:

a) Preventive maintenance and periodic visual inspection requirements and

b) Integrated leak test requirements for each system in accordance with the Surveillance Frequency Control Program.

In the LAR, the licensee stated that this program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Core Spray, High Pressure Coolant Injection, Residual Heat Removal, Reactor Core Isolation Cooling, Reactor Water Sampling, Post Accident Sampling, Reactor Water Cleanup, Hydrogen Recombiners, Primary Containment Monitoring, Control Rod Drive discharge headers, and Standby Gas Treatment.

The licensee reviewed the program test history and identified no failures of the TS functions that would have been solely detected by the periodic performance of this program. The licensee stated that the impact, if any, on system availability is minimal from the proposed interval change from 18 to 24 months. The licensee determined the impact of this change on safety, if any, is small.

The NRC staff reviewed the proposed change and determined that the proposed change for the TS 5.5.2 program is consistent with the intent of GL 91-04. The NRC staff finds the proposed interval extensions for the above program acceptable because: (1) the effect on safety would be small, (2) historical data do not contradict this conclusion, and (3) no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

3.2.2.16 TS 5.5.8, “Ventilation Filter Testing Program (VFTP)”

The licensee requests a revision of TS 5.5.8 that increases the testing interval from 18 months to 24 months, for a maximum interval of 30 months, including the 25 percent extension afforded by TS SR 3.0.2 for the below listed items within TS 5.5.8.

5.5.8.a Demonstrate for ventilation systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% when tested at the system flowrate specified below ± 10%.

5.5.8.b Demonstrate for ventilation systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1% when tested in accordance with ASTM D3803-1989 at the system flowrate specified below ± 10%.

5.5.8.c Demonstrate for Control Room Emergency Ventilation System that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C [degrees Celsius] (86°F) and the relative humidity of 95%.

5.5.8.e Verify by a visual inspection the absence of foreign materials and gasket deterioration.

The VFTP requires testing of the ESF filter ventilation systems in accordance with RG 1.52, “Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants,” Revision 2 (Reference 16), and ASTM D3803-1989 at the frequencies specified in RG 1.52, Revision 2 (i.e., 18 months). The performance/evaluation criteria can be found in TS 5.5.8. The proposed changes would increase the frequencies from 18 to 24 months by satisfying the requirements in paragraphs a, b, c, and e without any change to those paragraphs.

The licensee reviewed the VFTP test history and identified one failure of a single charcoal bed sample taken from the Control Room Emergency Ventilation System (CREVS) compensatory filter train. Based on the design of the CREVS and no other failures, the licensee stated that the impact, if any, on system availability would be minimal from the proposed interval change from 18 to 24 months. The licensee determined that the impact of this change on safety, if any, is small.

The NRC staff reviewed the proposed change and determined that the proposed change for the TS 5.5.8 program is consistent with the intent of GL 91-04. The NRC staff finds the proposed interval extensions for the above program acceptable because: (1) the effect on safety would be small, (2) historical data do not contradict this conclusion, and (3) no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

3.2.2.17 TS 5.5.16, “Surveillance Frequency Control Program”

In the LAR, the licensee proposed to extend SR frequencies from 18 to 24 months for numerous SRs in support of an extended fuel cycle. However, the justification cited by the licensee in its proposal is GL 91-04 and TS 5.5.16b states that changes to the frequencies listed in the SFCP shall be made in accordance with NEI 04-10, Revision 1.

By RAI dated September 17, 2024 (Reference 17), the NRC staff brought this issue to the licensee’s attention, stating:

> Justify the accuracy of current TS 5.5.1[6]b given that the proposed changes in the LAR are being supported by a guidance document not currently listed in the SFCP TS.

In its response dated October 31, 2024 (Reference 4), the licensee proposed to revise TS 5.5.16b as follows (additions shown in bold underlined text):

> Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, “Risk-Informed Method for Control of Surveillance Frequencies,” Revision 1 **<u>or as specifically approved by the NRC</u>**.

The NRC staff reviewed the supplemented language and finds that the proposed change to TS 5.5.16b is acceptable because it appropriately allows changes to the frequencies in the SFCP as approved by the NRC and not just in accordance with NEI 04-10.

3.2.2.18 Non-Calibration SR Interval Change Evaluation Conclusion

Based upon the NRC staff's review of the LAR, as supplemented, which is summarized above, the proposed changes meet the guidance of GL 91-04. The NRC staff concludes that the proposed interval extensions for the non-calibration SRs are acceptable because: (1) the effect on safety would be small, (2) historical data do not contradict this conclusion, and (3) no assumptions in the plant licensing basis would be invalidated as a result of the proposed changes.

3.3 <u>Calibration Changes</u>

3.3.1 GL 91-04 Guidance and Licensee Evaluation

GL 91-04 identifies seven steps to evaluate calibration changes. The licensee provided the following general evaluations of those seven steps.

> **<u>STEP 1</u>:** Confirm that instrument drift as determined by as-found and as-left calibration data from surveillance and maintenance records has not, except on rare occasions, exceeded acceptable limits for a calibration interval.
>
> <u>EVALUATION</u>
>
> Historically, as-found tolerances used in Surveillance Procedures at Turkey Point have been based on instrument accuracy. As required by plant procedures, out of tolerance conditions detected during performance of surveillance procedures are entered into the Corrective Action Program (CAP) for evaluation and trending. This ensures identification of occurrences of instruments found outside of their specified tolerances and instruments whose performance is not as anticipated by the setpoint analysis. When an instrument under surveillance is found to have exceeded the as-found tolerance (i.e., acceptable limits) provided in the surveillance procedures, a CAP report is initiated and referenced/attached to the Work Order and an operability determination is performed to determine if the out of tolerance condition has challenged the operability of the loop.
>
> The difference between as-found and as-left data collected during performance of surveillance procedures represents the combined effects of instrument reference accuracy, calibration error, time dependent error and normal radiation effects. Statistical analysis was performed on the as-found and as-left data from surveillance procedures for all instruments which perform a safety related (SR) function to determine a statistical drift value that is representative of data collected from the prior four cycles of plant operation or since when the instrument was replaced. The statistically determined drift was extrapolated for a surveillance interval of 30 months (24 months plus 25 [percent]). The 30-month extrapolated drift values were used to determine the impact on loop uncertainty, Trip Setpoint, and Allowable Value for the instrumentation providing SR functions listed in Table C-1 ["Applicable Instrumentation," of the licensee's Generic Letter

91-04 Evaluation document]. The Allowable Values and Trip Setpoints were either confirmed to be conservative assuming a 30-month surveillance interval, or identified for revision as shown in Table C-3 ["Summary of Allowable Value and Trip Setpoint Changes," of the licensee's Generic Letter 91-04 Evaluation document]. The availability of margin between the actual plant setting and the Trip Setpoints was confirmed and/or adjusted to assure that the existing as-found setting tolerance specified in surveillance procedures do not challenge the Allowable Values as specified in the Technical Specifications or as modified as a part of this amendment request.

STEP 2: Confirm that the values of drift for each instrument type (make, model, and range) and application have been determined with a high probability and a high degree of confidence. Provide a summary of the methodology and assumptions used to determine the rate of instrument drift with time based upon historical plant calibration data.

EVALUATION

A listing of the lead instrument make, model, and range affected by this submittal is provided in Table C-1 ["Applicable Instrumentation," of the licensee's Generic Letter 91-04 Evaluation document]. The effect of longer calibration intervals on the TS instrumentation is evaluated by performing an instrument drift study. A discussion of the drift analysis performed for Turkey Point Units 3 and 4 is provided in Section C.1 [of the LAR]. Based on the drift results summarized in Section C.1 and Table C-2 ["Statistical 95/95 Calibration and Drift Values for 24-Month Cycles," of the licensee's Generic Letter 91-04 Evaluation document] it was concluded that all the listed functions can be extended from 18-month surveillance intervals to 24-month intervals (30-months including 25 [percent] allowance).

STEP 3: Confirm that the magnitude of instrument drift has been determined with a high probability and a high degree of confidence for a bounding calibration interval of 30 months for each instrument type (make, model number, and range) and application that performs a safety function. Provide a list of the channels by TS section that identifies these instrument applications.

EVALUATION

In accordance with the methodology described above, the magnitude of instrument drift has been determined with a high probability and a high degree of confidence (typically 95/95) for a bounding calibration interval of 30 months for each instrument make, model, and range. The proposed allowance to apply 1.25 grace to SRs with frequency of 24 months is based on this approach. The list of affected channels by TS section, including instrument make, model, and range, is provided in Table C-1 of [the LAR].

STEP 4: Confirm that a comparison of the projected instrument drift errors has been made with the values of drift used in the setpoint analysis. If this results in revised setpoints to accommodate larger drift errors, provide proposed TS changes to update trip setpoints. If the drift errors result in revised safety analysis to support existing setpoints, provide a summary of the updated analysis conclusions to confirm that safety limits and safety analysis assumptions are not exceeded.

EVALUATION

The impact of instrument drift was evaluated for the proposed calibration SR interval changes. As a result of the drift evaluation, Turkey Point Units 3 and 4 instrumentation setpoint and uncertainty calculations will be revised, as necessary, to reflect the proposed calibration SR interval changes. The evaluation also resulted in proposed changes to TS Allowable Values and a Trip Setpoint. The affected calibration surveillance procedures will be revised as part of implementation.

STEP 5: Confirm that the projected instrument errors caused by drift are acceptable for control of plant parameters to effect a safe shutdown with the associated instrumentation.

EVALUATION

The previous steps discuss the evaluation of the impact of drift on instrument setpoint and uncertainty calculations associated with surveillance intervals of 30 months (24 months plus 25 [percent]). The evaluation includes the control system [the NRC staff interprets the licensee to be referring to the protection systems (e.g., reactor trip and ESFAS) and not the broader term “control system”] instrumentation used for safe shutdown. The evaluation provides assurance that the control system instrumentation will perform with the required accuracy to affect a safe shutdown.

STEP 6: Confirm that all conditions and assumptions of the setpoint and safety analyses have been checked and are appropriately reflected in the acceptance criteria of plant SR procedures for channel checks, channel functional tests, and channel calibrations.

EVALUATION

The affected calibration surveillance procedures will be revised as part of implementation, prior to the first application of the SRs in the Turkey Point STI [surveillance test interval] with a frequency of 24 months. Existing plant processes ensure that the conditions and assumptions of the setpoint and safety analyses have been checked and are appropriately reflected in the acceptance criteria of plant surveillance procedures for channel checks, channel functional tests and channel calibrations.

STEP 7: Provide a summary description of the program for monitoring and assessing the effects of increased calibration surveillance intervals on instrument drift and its effect on safety.

EVALUATION

> Instruments with TS calibration SR intervals of up to 30 months will be monitored and trended in accordance with station procedures including recording of as-found and as-left calibration data. As required by plant procedures, out of tolerance conditions are entered into the corrective action program and are evaluated and trended. This approach will identify occurrences of instruments found outside of their allowable value and instruments whose performance is not as assumed in the drift or setpoint analysis. When the as-found conditions are outside the as-found tolerance (i.e., acceptable limits), an evaluation will be performed in accordance with the station corrective action program to evaluate the effect, if any, on plant safety.

3.3.2 NRC Staff Evaluation of Calibration SR Interval Changes in Accordance with GL 91-04 Criteria

The NRC staff finds the explanation of the seven steps above acceptable per the guidance of GL 91-04. The justification for each SR is explained further below.

Step 1 Evaluation

In its evaluation of Step 1, the licensee provided an explanation of its maintenance of its nominal trip setpoints and instrument accuracy and how values found outside the accepted tolerance would be managed via its CAP. In addition, in the Background Section of its Generic Letter 91-04 Evaluation, it states, in part:

> Turkey Point Units 3 & 4 historical SR performance data and associated maintenance records were reviewed to evaluate the effect of these changes on safety. The Surveillance Failure Analysis (SFA) included non-calibration SRs, calibration SRs with setpoints (TS Allowable Values), and calibration SRs without setpoints (no TS Allowable Values)…. The SFA identified no SR failures that would call into question the acceptability of the proposed extension of SR intervals.

The NRC staff's understanding of the licensee's last statement above regarding the acceptability of the proposed extension of SR intervals to mean, "the proposal to extend SR intervals." On this basis the staff notes that the licensee's analysis of historical as-found and as-left data did not identify any significant reasons to question the performance capability of its TS-related instrument channels to meet extended surveillance intervals. The staff also notes that the licensee's existing program for entering into a CAP process for evaluating individual surveillance results will serve to ensure that future out-of-tolerance surveillance results will alert licensee staff to the need for further evaluation. The staff also evaluated the licensee's accounting for drift based on historical performance and conclusions drawn from its SFA and found them to be acceptable because the methods applied were consistent with RG 1.105 guidance and WCAP-17070-P and WCAP-17504, whose methods have been applied in WCAP-18888-P and have been found to be acceptable.

Step 2 Evaluation

The licensee conducted its drift analysis study by reviewing its as-found and as-left measured values for a minimum of the last 3 refueling cycles to ensure a high degree of confidence in its results. This was confirmed by the licensee in its response to EICB-RAI-2. The evaluation method of a performance-based drift value, (i.e., AFT – ALT) is consistent with the approach outlined in WCAP-17504, which was approved by the NRC in October 2016 (ML16314A091), and is, therefore, acceptable.

Step 3 Evaluation

The licensee provided a list in its GL 91-04 evaluation of the make, model number, and range of the instruments associated with TS-controlled setpoints. The NRC staff independently evaluated several examples of the application of the licensee's calculations used in its setpoint methodology and found them to be consistent with NRC-approved guidance in RG 1.105 for combining instrument accuracy, drift, and calibration accuracy and, therefore, acceptable.

Step 4 Evaluation

With respect to its evaluation of Step 1, the licensee stated, in part:

> Statistical analysis was performed on the as-found and as-left data from surveillance procedures for all instruments which perform a safety related (SR) function to determine a statistical drift value that is representative of data collected from the prior four cycles of plant operation or since when the instrument was replaced. The statistically determined drift was extrapolated for a surveillance interval of 30 months (24 months plus 25 [percent]). The 30-month extrapolated drift values were used to determine the impact on loop uncertainty, Trip Setpoint, and Allowable Value for the instrumentation providing SR functions listed in Table C-1 [of the licensee's Generic Letter 91-04 Evaluation document]. The Allowable Values and Trip Setpoints were either confirmed to be conservative assuming a 30-month surveillance interval, or identified for revision as shown in Table C-3 [of the licensee's Generic Letter 91-04 Evaluation document]. The availability of margin between the actual plant setting and the Trip Setpoints was confirmed and/or adjusted to assure that the existing as-found setting tolerance specified in surveillance procedures do not challenge the Allowable Values as specified in the Technical Specifications or as modified as a part of this amendment request.

Using statistical analysis and data from the last four refueling cycles, the instrument channel drift data (i.e., drift error) was evaluated by the licensee and statistically extrapolated to a maximum surveillance interval of 30 months (24 months x 1.25 grace period). The incorporation of an updated unified setpoint methodology for TS-controlled setpoints has generally resulted in more conservative as-found and as-left tolerances and allowable values even after considering the longer updated drift error values associated with a longer surveillance interval due to a 24-month refueling cycle. The NRC staff notes that this was possible in some cases because some instruments had been replaced over the years with devices with improved accuracy and drift performance specifications. The NRC staff also notes that for those cases where the resulting AFTs and ALTs had been altered in the non-conservative direction, it was due to an expansion of the associated instrument span for the device.

Based on the licensee's evaluation of the projected instrument drift errors for a maximum of a 30-month surveillance interval, several changes were proposed to be made to some individual instrument channel's AVs and a trip setpoint (i.e., nominal trip setpoint). This evaluation also included changes to the nominal trip setpoint calculations associated with instrument channels' AVs and a trip setpoint (i.e., nominal trip setpoint) associated with the adoption of WCAP-18888-P. The methodology in WCAP-18888-P is consistent with the methodology in WCAP-17070-P (refer to Section 3.1 of this SE). WCAP-17070-P was previously evaluated by the NRC staff during its review and approval of the licensee's 2012 EPU. Its approach to establishing and maintaining nominal trip setpoints and AVs is also consistent with the approach in WCAP-17504. The staff determined the approach used in WCAP-18888-P is also consistent with NRC-approved guidance in RG 1.105 and is, therefore, acceptable.

Step 5 Evaluation

The drift analysis discusses the impact of drift on instrument setpoint and uncertainty calculations including instrumentation used for safe shutdown. The NRC staff evaluated the licensee's revised instrument "Drift Analysis Summary," Section C.1 in its Generic Letter 91-04 Evaluation in light of the proposed revisions to the NTSPs and AVs in the TS Tables.

10 CFR 50.36(c)(1)(ii)(A) states, in part, that "[l]imiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded." Based on the NRC staff's evaluation of "Drift Analysis Method," described in Section C.1of the licensee's Generic Letter 91-04 Evaluation and the approach used to calculate individual drift values that is described in WCAP-17504, the staff finds that, with the proposed changes, there would be sufficient margin between the NTSP and the AL to ensure that safety actions will be achieved (i.e., that the protective action will be achieved before the safety limit is exceeded), as is required by the NRC's regulations.

Step 6 Evaluation

The licensee stated that the impacted calibration surveillance procedures will be revised prior to the implementation of the revised SRs for the 24-month surveillance frequency. Additionally, conditions and assumptions in the setpoint safety analysis for existing plant procedures have been examined and have been appropriately reflected in surveillance test acceptance criteria.

The NRC staff understands the licensee's assurance that it will, consistent with its programs, incorporate the revised setpoint calculation and associated surveillance procedures during the implementation phase of this license amendment prior to the first application of the SRs in the Turkey Point STI with a frequency of 24 months. Elements of the NRC's inspection program may be used to verify the licensee's assertion that its surveillance procedures have been updated.

Step 7 Evaluation

The licensee described its management of as-found and as-left data when compared to acceptable limits. In its response to EICB-RAI-4, the licensee stated that when an as-found value is determined to be out of tolerance, an engineering evaluation will be performed in accordance with its plant procedures. This is consistent with TSTF-493, Option A, which uses footnotes in the TS to inform operators that when an as-found measured value is found to exceed the AFT, an engineering evaluation must be conducted.

The NRC staff determined that several methods used by the licensee support a multi-faceted approach to evaluate drift. These methods include:

- The licensee's new setpoint methodology (WCAP-18888-P),
- The licensee's surveillance procedures informing its staff to enter an out of tolerance TS controlled instrument channel into the licensee's CAP and to notify the control room operators, and
- The TS notes consistent with TSTF-493, Option A.

The NRC staff determined that the ongoing implementation of this approach will provide reasonable assurance that these programmatic attributes for monitoring and assessing the effects of increased calibration surveillance intervals on instrument drift and its effect on safety are adequate.

3.3.3 Category B - Calibration SRs Without Setpoints

This category consists of calibration SRs without setpoints (with no associated TS AVs).

3.3.3.1 TS 3.3.1, "Reactor Trip System (RTS) Instrumentation"

SR 3.3.1.9 Perform CHANNEL CALIBRATION – TS Table 3.3.1-1, Function 17 – Reactor Trip System Interlocks

The licensee's evaluation determined that the interlock test consists of verifying that the interlock is in its required state by observing the appropriate annunciator window. Additionally, the licensee stated that no failures of these TS functions would have been detected exclusively via the conduct of this SR based on the history of system performance related to the execution of this surveillance. The licensee also performed a confirmatory review that verified that its licensing basis functions are not impacted by the SR interval extension, which caused it to conclude that the impact on safety is small. Based on its review, the NRC staff concludes that the overall impact on safety is small and, therefore, the proposed change is acceptable.

The NRC staff notes that neutron detectors are excluded from SR 3.3.1.9.

3.3.3.2 TS 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation"

SR 3.3.2.6 Perform CHANNEL CALIBRATION – TS Table 3.3.2-1, Function 7 – ESFAS Interlocks

The licensee's evaluation of its test and calibration history related to SRs did not identify any failures of the TS functions that would have been detected solely by the periodic performance of this SR. The licensee also conducted a confirmatory review that verified that its licensing basis functions are not impacted by the SR interval extension. Although permissive and interlock setpoints allow the blocking of trips during plant startups and the restoration of trips when the permissive conditions are not satisfied, this is not explicitly modeled in the Safety Analyses.

Therefore, the licensee concluded that extending the surveillance does not negatively affect safety.

Based on its review, the NRC staff concludes that the overall impact on safety is small and, therefore, the proposed change is acceptable.

3.3.3.3 TS 3.3.3, "Post Accident Monitoring (PAM) Instrumentation"

SR 3.3.3.2 Perform CHANNEL CALIBRATION – TS Table 3.3.3-1, Functions 1-8 and 10-22

Note: Function 9 is a Valve Position Indicator for Penetration Flow Path Containment Isolation Valve and has no associated calibration function. The PAM CHANNEL CALIBRATION provides indication only, including alarm function, and does not include setpoints for control or actuation functions.

In accordance with GL 91-04, the licensee evaluated the test history for these functions and determined that no failures of TS functions would have been identified exclusively by the performance of these SRs. The licensee also performed a confirmatory review that verified that its licensing basis functions are not impacted by the SR interval extension, which caused it to conclude that the impact on safety is small. Based on its review, the NRC staff concludes that the overall impact on safety is small and, therefore, the proposed change is acceptable.

3.3.3.4 TS 3.3.4, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation"

SR 3.3.4.3 Perform CHANNEL CALIBRATION – TS Table 3.3.4-1, Function 1 – Control Room Air Intake Radiation Level

In accordance with GL 91-04, the licensee evaluated the test history for these functions and determined that no failures of TS functions would have been identified exclusively by the performance of these SRs. The licensee also performed a confirmatory review that verified that its licensing basis functions are not impacted by the SR interval extension, which caused it to conclude that the impact on safety is small. Based on its review, the NRC staff concludes that the overall impact on safety is small and, therefore, the proposed change is acceptable.

3.3.3.5 TS 3.4.12, "Overpressure Mitigating Systems (OMS)"

SR 3.4.12.5 Perform CHANNEL CALIBRATION for each required PORV actuation channel.

In the LAR, the licensee stated that the technical justification for the proposed change addresses the information contained in GL 91-04. For calibration SRs without setpoints, GL 91-04 provides guidance to licensees on how to perform evaluations and confirmations as described in section 3.2.1 of this SE to support the change in surveillance intervals to accommodate a 24-month fuel cycle.

In the LAR, the licensee provided a summary of the evaluation to address the GL 91-04 guidance. Specifically, the licensee stated:

> The OMS controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR [Part] 50, Appendix G.
>
> SR 3.4.12.5 states 'Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.'

The licensee reviewed the SR test history and identified no failures of the TS functions that would have been solely detected by the periodic performance of this SR. The licensee stated that the impact, if any, on system availability is minimal from the proposed SR interval change from 18 to 24 months. The licensee determined that the impact of this change on safety, if any, is small.

The NRC staff reviewed the proposed change and determined that the proposed change for SR 3.4.12.5 is consistent with the intent of GL 91-04. The NRC staff finds the proposed interval extension for the above SR acceptable because: (1) the effect on safety would be small, (2) historical data do not contradict this conclusion, and (3) no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

3.3.3.6 TS 3.4.15, "RCS Leakage Detection Instrumentation"

> SR 3.4.15.3 Perform CHANNEL CALIBRATION of the required containment sump monitor.

The licensee's evaluation identified multiple failures in the Turkey Point Unit No. 4 level transmitter LT-4-6308A during the evaluation period. No failures were noted for Turkey Point Unit No. 3 components associated with this SR. Regarding this failure information the licensee stated, in part:

> A review of SR test history identified two failures of portions of the instrument loop for containment sump monitor LT 4 6308A. However, based on information provided by Turkey Point I&C Engineering the 'As Found' data for all control room indications associated with this loop were within their acceptance range in both cases. This includes non-safety/quality related indication via ERDADS and LI-4-6308A (RG 1.97 type B), as well as safety related recorder LR-4-6308A (RG 1.97 type A).

Based on the licensee's evaluation of the failures and its research into the extent of condition, as well as the conclusion that ultimately all "As Found" data for all control room indications for this loop were within the bounds of their acceptance criteria, the NRC staff determined that the historical test records, even with the failures, do not invalidate the licensee's claim that extending the surveillance interval would have a small impact on safety. Additionally, the licensee's confirmatory review of the licensing basis in its UFSAR indicates that the impact on safety would be small. Based on its review, the NRC staff concludes that the overall impact on safety is small and, therefore, the proposed change is acceptable.

SR 3.4.15.4 Perform CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor.

In accordance with GL 91-04, the licensee evaluated the test history for these functions and determined that no failures of TS functions would have been identified exclusively by the performance of this SR. The licensee also performed a confirmatory review that verified that its licensing basis functions are not impacted by the SR interval extension, which caused it to conclude that the impact on safety is small. Based on its review, the NRC staff concludes that the overall impact on safety is small and, therefore, the proposed change is acceptable.

3.3.4 Category C – Calibration SRs With Setpoints

This category consists of the following calibration SRs with setpoints (with associated TS AVs):

TS 3.3.1, "Reactor Trip System (RTS) Instrumentation"

SR 3.3.1.9 Perform CHANNEL CALIBRATION

For SR 3.3.1.9, the nuclear instrumentation channels, with the exception of the neutron detectors, are required to be subject to channel calibration.

SR 3.3.1.10 Perform CHANNEL CALIBRATION

TS 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation"

SR 3.3.2.6 Perform CHANNEL CALIBRATION

TS 3.3.6, "Containment Ventilation Isolation Instrumentation"

SR 3.3.6.3 Perform CHANNEL CALIBRATION

3.3.4.4 Category C – Calibration SRs With Setpoints Evaluation

In its response to RAI "EICB-RAI-2" (Reference 4), the licensee stated that the method to establish and maintain reactor trip and ESFAS nominal trip setpoints in WCAP-18888-P is consistent with the method in WCAP-17504, which was approved by the NRC in 2016 (ML16314A091). In Section 3.1.8, "Data Used to Select the Trip Setpoint," of the SE regarding WCAP-17504, the NRC staff noted, in part, that:

> For reactor trip and ESFAS initiation functions, the WSM [Westinghouse Setpoint Methodology] evaluation of drift is based on a two-sided (±) 95 percent probability at a 95 percent confidence level. A significant volume of as-left and as-found data is collected over a minimum of **[[ ]]** to verify the magnitude of drift remains bounded along with reference accuracy and calibration accuracy in the allowance for AFT.

Additionally, the NRC staff noted the steps and/or approaches the licensee will take to evaluate the outlying data and to ensure that the data is properly evaluated and reconciled in a manner such that the contributing terms to the uncertainties associated with the overall channel statistical accuracy (CSA) for the given instrument channel will be applied to the overall CSA calculation. This method is also described in Section C.1 of the licensee's Generic Letter 91-04 Evaluation. This approach, described in both references, is applied to ensure that each instrument channel setpoint has been developed to ensure that a two-sided 95/95 tolerance interval value for setpoints can be satisfactorily determined.

Related to setpoint uncertainty analysis, the setpoint values (i.e., nominal setpoint values) from WCAP-12745 were updated to align with the method used in WCAP-18888-P.

The licensee stated that the AVs for the Turkey Point TS are "performance based" and are determined by adding (or subtracting) the rack calibration accuracy (RCA=ALT) of the device(s) tested during the Channel Operational Test to the Trip Setpoint in the non-conservative direction (i.e., toward or closer to the Safety Analysis Limit) for the application. This is consistent with the previously approved methodology in WCAP-17504-P.

In addition, the licensee stated that the AV used to determine operability of a setpoint function in the TS is not a function of sensor drift. Typically, the AVs are calculated by comparing the NTSP (i.e., the LSSS value per 10 CFR 50.36) and the Rack Calibration Accuracy (RCA) in accordance with the methodology described in Section 4 of WCAP-18888-P. The proposed AV changes in Table C-3 of the licensee's Generic Letter 91-04 Evaluation changed primarily due to different computational methods from the WCAP-18888-P methodology when compared to the method used in WCAP-12745-P, since WCAP-12745-P used a different method for combining statistical terms.

Concerning the change of the AV for ESFAS Function 1f, this function is implemented within the Eagle-21 digital protection system. The input card that implements ESFAS Function 1f was designed to accommodate up to a 0-150°F span. The Eagle-21 racks are calibrated to operate on a 0-100°F span. The output of the Eagle-21 racks are distributed to the plant systems, rod control, indicators, etc., which are based on the original plant as-built configuration with a 0-75°F span. Thus, the existing plant hardware did not have to be replaced to accommodate a larger 0-100°F span when the Eagle-21 racks were installed, and the increased span resulted in uncertainties that could still be accommodated within the total instrument channel uncertainty calculations.

The licensee explained that the relaxation of the AV for ESFAS Function 1f is not due to a change in setpoint uncertainty methodology, but instead is due to the use of the 100°F span over the previous conservative use of the 75°F span as the coincident logic is based on direct output from the Eagle-21 derivation and not from the 75°F output to the control and indication functions. This approach is acceptable because the outcome of this range adjustment results in no change to the NTSP and the raising of the AV results in a more conservative acceptance limit.

The NRC staff reviewed a representative sample of the calculation summaries presented in WCAP-18888-P, which provided confidence that no alteration to any of the other affected setpoints' AVs was required. Additionally, as the two-sided as-found and as-left tolerance values, which are equivalent in magnitude to the one-sided AV, are performance-based versus directly tied to drift values and any out of tolerance as-found measured values will be assessed via an engineering evaluation to determine the extent of the out of tolerance condition, it is consistent with the approach in TSTF-493.

Based on the NRC staff's review of the method applied to establish and maintain TS controlled instrument channel setpoints using WCAP-18888-P, the NRC staff determined that the SRSS methodology used to calculate the proposed trip setpoints and AVs was consistent with ANSI/ISA 67.04.01. In particular, the method applied in the licensee's NTSP agrees with the "Combination of Uncertainties (SRSS method)" of Part I of ANSI/ISA 67.04.01-2018 and RG 1.105, Revision 4. It therefore provides reasonable assurance that the impacted setpoints were evaluated adequately. Equations used in the calculations within the summary report are therefore consistent with the guidance in RG 1.105.

The NRC staff also determined that the method to establish and maintain setpoints for TS controlled instrument channels used by the licensee in WCAP-17070-P and WCAP-17504-P and adopted in WCAP-1888-P adequately addresses the associated drift term for the functions described in the LAR. This is based on the adoption of TSTF-493 Option A where it describes that should any as-found measured value be found outside of its as-found tolerance limit (which in this case is the AV in the non-conservative direction), then an engineering evaluation shall be conducted to determine the cause of the malfunction before the channel is returned to service.

Therefore, the NRC staff finds that the explanations of setpoint establishment and maintenance of setpoints provided in WCAP-18888-P, along with the TSTF-493 Option A footnotes and the licensee's evaluation in its Generic Letter 91-04 Evaluation describing the seven acceptable steps per the guidance of GL 91-04, adequately addresses the criteria in GL 91-04.

3.3.4.5 Category C – Calibration SRs With Setpoints Review Conclusion

The NRC staff's review considered the setpoint methodology in WCAP-18888-P that applied the SRSS method to the development of setpoints and AVs, which is consistent with the approach outlined in ANSI/ISA 67.04.01 that employs the use of an AFT and ALT. The use of these terms (i.e., AFT and ALT) and the manner by which they are implemented in the licensee's TS aligns with TSTF-493, Option A. Related to the conduct of an engineering evaluation should an instrument channel's AFT be found out of tolerance, plant procedures provide reasonable assurance that, should that condition occur, an engineering evaluation will be conducted before restoring the channel to service.

The WCAP-18888-P approach is consistent with the method described in WCAP-17504-P, which uses a performance-based approach in the development of its AV, rather than a direct tie to the rack portion of the "drift" term. Additionally, the NRC staff's evaluation of the licensee's approach to meet the seven step criteria of GL 91-04 determined that the method applied provides reasonable assurance that the establishment and maintenance of instrument channel setpoints and AVs is acceptable.

Therefore, on the basis of the above determinations, the NRC staff concludes that the licensee-proposed application of the extended surveillance intervals for TS-controlled instrument channel setpoints is acceptable.

3.3.5 SR Extension Technical Conclusion

The licensee provided an explanation for calibration related SR changes and for non-calibration related SR changes. The licensee's analyses for these SR changes were reviewed and found acceptable by NRC staff because its GL 91-04 evaluation supports the determination that the effect on plant safety associated with the proposed SR interval changes from 18 to 24 months, if any, is small. The justification for previously documented failures not having an effect on safety if the proposed SR interval were to be adopted was found to be consistent with the guidance of GL 91-04. The failure report was reviewed by the staff and the failure report data were found to be acceptable by the staff per the guidance of GL 91-04. Based on the above, the NRC staff finds that the proposed extension of surveillance intervals from 18 to 24 months is acceptable and that the changes proposed in the LAR continue to meet 10 CFR 50.36(c)(5) by providing administrative controls necessary to assure operation of the facility in a safe manner.

3.4 Evaluation of Proposed Advanced Fuel Features

The key features of the current fuel and the fuel proposed to be used in the 24-month cycles are summarized in table 1 below.

Table 1: Key Features of Current and Proposed Fuel

Feature	Current Fuel	Proposed Fuel
Cladding Material	Optimized ZIRLO	AXIOM No changes to cladding dimensions.
Fuel Pellet Composition	UO_2	ADOPT (UO_2 doped with small amounts of chromia and alumina). Higher density, no changes to pellet dimensions.
Burnable Absorbers	IFBA [Integral Fuel Burnable Absorber] coated UO_2	IFBA coated ADOPT fuel. Gadolinia doped UO_2 fuel.
Mid and IFM [Intermediate Flow Mixer] Grid Material	ZIRLO	PRIME Low Tin ZIRLO. No changes to Mid and IFM grid dimensions.
Bottom Nozzle	DFBN [Debris Filter Bottom Nozzle]	PRIME Bottom Nozzle: ADFBN-LP [Low Pressure Advanced DFBN], Adapter Plate Flow Hole Geometry Change, **[[ ]]** lower loss coefficient than DFBN and **[[ ]]** overall fuel assembly loss coefficient.

3.4.1 Nuclear Core Design

This section evaluates the Turkey Point core design analysis to verify the cycle-specific design limits and the key safety parameters used in the reload analysis. Turkey Point currently uses Westinghouse 15 Upgrade fuel product with the IFBA. The effects on the nuclear design bases and methodologies due to a transition from the current fuel to the PRIME fuel design with AXIOM fuel cladding, ADOPT fuel pellets, and the introduction of the burnable absorber Gadolinia (GAD) are evaluated in this section. This section also evaluates the impact of a transition from the current 18-month cycle design to a 24-month cycle design with new fuel on the nuclear design for Turkey Point.

3.4.1.1 Nuclear Core Design Analysis

The Turkey Point core design is based on NRC-approved methods described in WCAP-9272-P-A, Revision 0, "Westinghouse Reload Safety Evaluation Methodology" (Reference 19). The licensee plans to transition Turkey Point from the current 18-month cycle design to a 24-month cycle design with new fuel within two transition cycles. The licensee analyzed specific values for loading pattern dependent safety parameters for the transition cores as well as the 24-month equilibrium core. The licensee analyzed key safety parameters of power distribution, peaking factors, rod worths, and reactivity coefficients and kinetic parameters to ensure that margin to existing limits remains for the transition cores as well as the 24-month equilibrium core. The use of AXIOM fuel cladding has been approved by the NRC in WCAP-18546-NP-A, Revision 0 (Reference 20), and the use of ADOPT fuel pellets has been approved by the NRC in WCAP-18482-NP-A, Revision 0 (Reference 21).

The licensee stated in the LAR that to meet the energy requirements of a 24-month cycle, the first transition cycle will entail loading 81 new fuel assemblies of the upgraded PRIME fuel design with AXIOM fuel cladding and ADOPT fuel pellets. The subsequent loadings will follow alternating feed patterns of 76 or 77 assemblies. The licensee stated that for the analysis performed, each feed region was fully enriched to 4.95 weight percent, including fully enriched blankets with IFBA utilized in tandem with the ADOPT fuel pellets.

These core designs were evaluated by the licensee in order to show that sufficient margin exists between the key safety parameter values and their corresponding limits. This allows for flexibility in the development of reload cores. The key parameters evaluated by the licensee in the LAR were physical core characteristics, shutdown margin and maximum boron concentration, moderator temperature coefficient (MTC), Doppler coefficient, beta effective, trip reactivity, axial power distribution, power distribution, and peaking factors. These parameters were evaluated for the following events: boron dilution, rod withdrawal and dropped RCCA, locked rotor, steamline break and feedwater malfunction, anticipated transient without scram, and inadvertent loading of an assembly.

3.4.1.2 Results

The licensee analyzed shutdown margin and boron concentration since these parameters are loading pattern dependent. The licensee stated in its evaluation that Turkey Point core designs have a requirement of 1770 per cent mille (pcm) shutdown margin, which is a function of the power defect from full power to hot zero power (HZP) at the time of a trip and the type of fuel that is placed under control rod locations. The licensee further stated that due to the aggressive nature of the 24-month cycle designs, the 1770 pcm shutdown margin could not be met and proposed a new shutdown margin limit of 1700 pcm. In response to NRC staff questions, the licensee stated in a letter dated October 3, 2024 (Reference 3), that an analysis performed by the Westinghouse nuclear design group determined that the new shutdown margin value developed using the Westinghouse conservative methodology allows sufficient margin to accommodate any anticipated variance in cycle-to-cycle changes. The NRC staff reviewed the technical justification presented and found it to be acceptable since it uses conservative analysis methods and showed margin to accommodate cycle-to-cycle variations.

The licensee stated that loading patterns developed for the transition and equilibrium cores continued to meet the safety parameter MTC limits. The licensee stated that the existing Doppler coefficient limits were found to bound the 24-month cycles except for the beginning of cycle (BOC) least-negative Doppler-only power defect. The licensee reevaluated the condition to regain margin for core design to support the 24-month cycles. The licensee evaluated effect of radial power distribution due to the 24-month cycle loading patterns and found the change to be within normal cycle-to-cycle variations when compared to the 18-month cycle. The boron dilution accident was re-evaluated as part of the non-LOCA transients using the new 1700 pcm shutdown margin and was found to be within the requirements. The licensee analyzed the dropped rod event, with a maximum dropped rod worth of 1000 pcm, to ensure that the nuclear enthalpy rise limit was not exceeded in transition and equilibrium cores. The licensee-performed analysis of the hot full power (HFP) streamline break (SLB) event was for the 24-month cycles and was shown to be bounded by the safety analyses of record for this event with margin to the kw/ft centerline melt criteria. Further, margin to departure from nucleate boiling (DNB) was shown during a locked rotor event. The licensee analysis showed that the transition and equilibrium cycles meet the MTC criteria for the anticipated transient without scram (ATWS) event.

The NRC staff reviewed the details of the Turkey Point core design during the transition from co-resident fuel to the new fuel design and determined that the key safety parameters and peaking factors are maintained within their specified limits with margins to accommodate implementation of cycle-to-cycle variations. The NRC staff confirmed that cycle checks were performed against the UFSAR safety analyses. The NRC staff determined that the core design for the transition has been performed according to the NRC-approved methodology. The NRC staff notes that these findings are based on the nominal core designs discussed in the LAR, which demonstrate that the licensee can meet its design requirements using the proposed new fuel.

3.4.1.3 Compliance with NRC Imposed Limitations and Conditions

The NRC staff evaluation of compliance with the limitations and conditions (L&Cs) listed in the SEs of the topical reports (TRs) used in the Nuclear Design events analysis is given below. This evaluation is based on the staff's review of Reference 1, Enclosure 2.

<u>WCAP-9272-P-A</u>

No limitations and conditions applicable to nuclear design are listed in the SE of this TR.

<u>WCAP-18546-NP-A</u>

<u>L&C #1</u>

> AXIOM cladding must be used with the NRC-approved PWR designs.

<u>Compliance with L&C #1</u>

Turkey Point, Unit Nos. 3 and 4 are Westinghouse licensed PWR designs.

<u>L&C #2</u>

> AXIOM cladding must be used with the NRC-approved Westinghouse and CE [Combustion Engineering] fuel designs with corresponding pellet and assembly dimensions.

Compliance with L&C#2

The licensee stated that AXIOM cladding for the 15x15 Upgrade fuel was licensed for Turkey Point, Unit Nos. 3 and 4 through the Fuel Criteria Evaluation Process (FCEP), which was approved for use by the NRC in WCAP-12488-A.

L&C #3

> AXIOM cladding must be used with the NRC-approved fuel materials and pellet coatings or additives (e.g., ADOPT, IFBA, gadolinium).

Compliance with L&C #3

The licensee stated that the AXIOM cladding will be used with the NRC-approved ADOPT fuel pellets with IFBA coatings. The licensee further stated that the ADOPT fuel additives and their use with IFBA coatings are approved for use in WCAP-18482-NP-A.

L&C #4

> Currently fuel burnup shall be limited to 62 GWd/MTU peak rod average for all cladding types, however, fuel rod burnup **[[ ]]** may be allowed once additional information specific to burnup to **[[ ]]** is submitted and approved by the NRC.

Compliance with L&C#4

The licensee stated that a peak rod average burnup limit of 62,000 MWD/MTU (Fuel Assembly burnup of approximately 56,000 MWD/MTU) will be applied.

L&C #5

> Best Estimate Oxide Thickness < 100 μm [micrometers].

Compliance with L&C #5

The licensee stated that the measured maximum oxide thickness of the AXIOM alloys are less than 50 μm for a burnup of close to 75 GWd/MTU. The licensee further stated that the best estimate oxide thickness will be less than the allowed 100 μm for a peak rod average burnup of 62,000 MWD/MTU.

L&C #6

> Best Estimate hydrogen pickup (HPU) ≤ **[[ ]]**.

Compliance with L&C #6

The licensee stated that the overall maximum hydrogen content for AXIOM cladding is ~ 200 ppm due to the combination of low maximum oxide thickness and low HPU ratio.

WCAP-18482-NP-A

L&C #1

> Licensees must demonstrate that the CRE [control rod ejection] analytical models, methods, and acceptance criteria are applicable to fuel designs containing ADOPT fuel pellets and capture all relevant fuel burnup and cladding corrosion related phenomena….

Compliance with L&C#1

The licensee used NRC-approved methodology from WCAP-15806-NP-A, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics" (Reference 36), for the Turkey Point CRE analysis with ADOPT fuel and followed the guidance provided in RG 1.236 (Reference 50) for the generation of CRE acceptance criteria with regard to ADOPT fuel pellets relating to all fuel burnup and cladding corrosion related phenomena using ADOPT burnup dependent limit values.

L&C #2

> ADOPT fuel must be used with the NRC-approved Westinghouse and CE PWR designs.

Compliance with L&C #2

Turkey Point, Unit Nos. 3 and 4 are Westinghouse licensed PWR designs.

L&C #3

> ADOPT fuel must be used with the NRC-approved Westinghouse and CE fuel designs with corresponding pellet and assembly dimensions.

Compliance with L&C #3

The licensee stated that the ADOPT fuel will replace standard uranium dioxide (UO2) pellets used with the Westinghouse 15x15 Upgrade PRIME fuel design. There are no changes to the corresponding pellet and assembly dimensions as compared to the existing licensed fuel design. The 15x15 Upgrade fuel design was licensed for Turkey Point, Unit Nos. 3 and 4 through the FCEP, which was approved for use by the NRC.

L&C #4

> ADOPT fuel shall be used with the NRC-approved zirconium based cladding materials, such as ZIRLO® and Optimized ZIRLO™.

Compliance with L&C #4

The licensee stated that ADOPT fuel will be used with AXIOM cladding material, which is approved for licensing applications in WCAP-18546-NP-A.

L&C #5

> ADOPT fuel may be used with or without annular pellets and application of ZrB2 integral fuel burnable absorber (IFBA) coating but must be used consistent with the defined IFBA parameters in applicable NRC-approved fuel performance or product TRs.

Compliance with L&C#5

The licensee stated that ADOPT fuel will be used with annular blankets and the application of integral burnable absorber coating consistent with NRC-approved TR WCAP-17642-NP-A, Revision 1, “Westinghouse Performance Analysis and Design Model (PAD5)” (Reference 22).

L&C #6

> Fuel burnup shall be limited to 62 GWd/MTU peak rod average for all cladding types.

Compliance with L&C #6

The licensee confirmed that the peak rod average burnup limit of 62,000 MWD/MTU is applied and will not exceed the maximum allowable during the reload evaluation process.

L&C #7

> Nominal pellet density range will be **[[** **]]**.

Compliance with L&C #7

The licensee stated that the ADOPT pellet density requirement is specified on the Westinghouse pellet drawing and is consistent with the value in the current TR WCAP-18482-NP-A (Reference 21).

L&C #8

> Fuel grain size range will be **[[** **]]** as measured according to ASTM E112 as linear intercept without correction factor, which corresponds to **[[** **]]** with correction.

Compliance with L&C #8

The licensee stated that the ADOPT fuel grain size range requirement is identified in the Westinghouse ADOPT pellet specification and is consistent with the current TR WCAP-18482-NP-A. The licensee also stated that the grain size is measured by the ASTM E112 linear intercept method without correction factor and that the average grain size is used as a product acceptance criterion for all pellets.

L&C #9

> Cr [Chromium] range from **[[ ]]** which corresponds to inclusion of Cr_2O_3 ranging from **[[ ]]**.

Compliance with L&C #9

The licensee stated that the chromium content requirement is identified in the Westinghouse ADOPT pellet specification and is consistent with the current TR WCAP-18482-NP-A.

L&C #10

> Al [Aluminum] ranging from **[[ ]]** which corresponds to inclusion of Al_2O_3 ranging from **[[ ]]**.

Compliance with L&C #10

The licensee stated that the aluminum content requirement is identified in the Westinghouse ADOPT pellet specification and is consistent with the current TR WCAP-18482-NP-A.

Based on its review of the LAR, the NRC staff finds that all the applicable limitations and conditions from the TRs listed in the nuclear design evaluation have been met.

3.4.1.4 Conclusion

Based on its review of the technical information provided by the licensee, which showed that there is sufficient margin to accommodate variations of key design and safety parameters during the transition to the 24-month cycle along with the new fuel design and showed compliance to the limitations and conditions from applicable topical reports, the NRC staff finds the nuclear core design analysis in the proposed request to be acceptable.

3.4.2 Core Thermal Hydraulic Design and Analysis

This section describes the T-H design and analyses that support the transition to the PRIME fuel design with AXIOM fuel cladding, ADOPT fuel pellets, and the introduction of the burnable absorber GAD at Turkey Point. The input parameters are from design documents, fuel assembly and component characteristics established by thermal hydraulic testing, and plant parameters provided by the licensee. The thermal hydraulic design of the core is established based on the following acceptance criteria in SRP section 4.4:

- There is at least a 95 percent probability at a 95 percent confidence level that the hot fuel rod in the core does not experience DNB during Condition I or II events.

- There is at least a 95 percent probability at a 95 percent confidence level that the hot fuel rod in the core does not melt during Condition I or II events.

Analytical assurance that DNB will not occur is provided by showing the calculated DNB ratio (DNBR) to be higher than the 95/95 design limit DNBR for Condition I and II events. The assurance that fuel centerline melt (FCM) will not occur is provided by comparing peak linear heat generation rate (PLHGR) to the linear heat generation rate (LHGR) corresponding to FCM. Assurance that fuel melting will not occur is provided by showing that the PLHGR is below the FCM limit for Condition I and II events. This evaluation was performed by the licensee, and the NRC staff confirmed that the licensee has acceptable margin for each cycle as part of the reload licensing process.

3.4.2.1 Thermal Hydraulic Design

Section 4.2 of the LAR provides core thermal hydraulic design while the thermal hydraulic analysis is presented in section 4.3 of the LAR. The licensee stated that the 15 Upgrade PRIME fuel assembly design incorporates material change of the mid and IFM grids from ZIRLO to Low Tin ZIRLO. The bottom nozzle flow hole geometries are specifically changed to reduce the pressure drop of the fuel assembly inlet region. In response to NRC staff questions, by letter dated October 3, 2024 (Reference 3), the licensee stated that the Upgrade PRIME bottom nozzle has approximately **[[ ]]** lower loss coefficient than the typical DFBN, which leads to an overall **[[ ]]** reduction for the overall fuel assembly loss coefficient. This change in flow loss between the existing and the proposed new fuel design will result in flow re-distribution which affects lift force and hold-down spring margin. The change also impacts the thermal performance in the thimble and dashpot regions of the fuel and generates cross flows in transition cores.

The licensee stated that the impact of this change will decrease as a larger number of PRIME assemblies are loaded in the core. The licensee determined that the proposed PRIME fuel assemblies would see a higher flow through them during transition cores due to low overall flow loss coefficient compared to current fuel. The licensee calculated lift and buoyancy forces to verify sufficient fuel assembly loads on the lower core plate for transition fuel as well as for a full core of new fuel. The licensee performed evaluation of the thimble and dashpot tubes with core components to confirm that the RCCA absorber rod in the 15x15 Upgrade PRIME fuel assembly will not experience bulk boiling in the thimble or surface boiling in the dashpot during the transition core as well as the full core for 24-month transition. The licensee verified that the design changes in the Upgrade PRIME fuel do not have any significant impact on the thimble bypass flow.

The NRC staff reviewed the thermal hydraulic design changes from existing fuel to the 15 Upgrade PRIME fuel assembly and found the changes to be acceptable given that they either have negligible impact on key safety parameters or have been appropriately analyzed to show sufficient margins to the applicable limits.

3.4.2.2 Thermal Hydraulic Analysis

For the thermal hydraulic analysis, the licensee stated that the current design basis for Turkey Point, Unit Nos. 3 and 4 includes the prevention of DNB on the limiting fuel rod with a 95 percent probability at a 95 percent confidence level (95/95). This design basis is documented in Turkey Point UFSAR, subsection 3.2.2, “Thermal and Hydraulic Design and Evaluation.” The 24-month cycle extension DNB analysis is based on this licensing basis.

The licensee stated that the DNB analysis of the fuel at Turkey Point is based on the Revised Thermal Design Procedure (RTDP) from NRC-approved TR WCAP-11397-P-A, “Revised Thermal Design Procedure” (Reference 23), and the WRB-1 DNB correlation (TR WCAP-8762-P-A, “New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux In Rod Bundles

With Mixing Vane Grids"(Reference 24)). The analysis is performed using the Westinghouse version of the VIPRE-01 subchannel analysis computer code (WCAP-14565-NP-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis" (Reference 25)). The licensee stated that for analyses which are outside of the range of applicability of the WRB-1 DNB correlation, the ABB-NV and WLOP correlations were used (WCAP-14565-NP-A). The ABB-NV correlation is for non-mixing vane fuel and the WLOP correlation is used to denote ABB-NV modification for low pressure.

The licensee performed thermal hydraulic analysis to support the 24-month cycle extension with new fuel for the following events: locked rotor, feedwater malfunction, RCCA drop/mis-operation, steam line break accident, and uncontrolled RCCA withdrawal from subcritical. By letter dated October 3, 2024 (Reference 3), the licensee provided analyses of these events and showed margin to the DNB design criterion for each of the events. The response also provided detailed comparison of the thermal hydraulic design parameters for the current design values as compared to the 24-month analysis values. The comparisons provided show that the only change in the key parameters was to the pressure drop across the core. In addition, the licensee performed evaluations to show that the core component cooling and flow stability design criteria are met for 24-month cycle extension conditions. The NRC staff evaluation of these events is in section 3.4.4 of this SE.

3.4.2.3 Results

The licensee stated that for each of the events analyzed, the thermal hydraulic analyses of the 24-month cycle extension for the transition core, as well as the full core of the new fuel, determined that the DNBR values are met with sufficient margin to the existing limits. The licensee stated that the cycle-specific evaluations in the future will be performed in accordance with the NRC-approved TR WCAP-9272-P-A.

3.4.2.4 Compliance with NRC Imposed Limitations and Conditions

The NRC staff evaluation of compliance with the L&Cs listed in the SEs of the TRs used in the thermal hydraulic design and analysis is given below. This evaluation is based on the staff's review of Reference 1, Enclosure 2.

<u>WCAP-9272-P-A</u>

No limitations and conditions applicable to thermal hydraulic design or analysis are listed in the SE of this TR.

<u>WCAP-8762-P-A</u>

No limitations and conditions applicable to thermal hydraulic design or analysis are listed in the SE of this TR.

<u>WCAP-11397-P-A</u>

By letter dated October 3, 2024 (Reference 3), the licensee stated that the design limit DNBR is calculated using the NRC-approved RTDP methodology which assures compliance with the previously discussed DNB criterion. This assures that the limitations and conditions in the topical report are met by the analysis performed.

WCAP-14565-NP-A

By letter dated October 3, 2024 (Reference 3), the licensee stated that the Westinghouse version of the VIPRE-01 (VIPRE) computer code used for DNB ratio calculations for the 24-month cycle extension analyses is in full compliance with the conditions specified in the SE of WCAP-14565-P-A for THINC and FACTRAN replacement.

Based on its review of the LAR, the NRC staff finds that all the applicable limitations and conditions from the topical reports listed in the thermal hydraulic design and analysis have been met.

3.4.2.5 Conclusion

Based on its review of the analyses provided by the licensee, which showed that there is sufficient margin to accommodate any variations in thermal hydraulics in the fuel designs during the transition to the 24-month cycle, the NRC staff finds the thermal hydraulic design evaluation to be acceptable. The NRC staff also finds that the licensee used appropriate NRC-methodology to perform the thermal hydraulic analysis and demonstrated compliance to limitations and conditions from applicable topical reports.

3.4.3 LOCA ECCS Analysis

The NRC regulations require that licensees of operating LWRs analyze a spectrum of accidents involving the LOCA to assure adequate core cooling under the most limiting set of postulated design basis conditions. The postulated spectrum of LOCAs range from scenarios with leakage rates just exceeding the capacity of normal makeup systems up through those involving rapid coolant loss from the complete severance of the largest pipe in the RCS. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core unless the water is replenished.

3.4.3.1 Small and Large Break LOCA Methodology

To support a transition to a 24-month fuel cycle, fuel transition to AXIOM cladding, ADOPT fuel pellets, and the PRIME fuel design, the licensee evaluated compliance to the ECCS acceptance criteria using the NRC-approved FULL SPECTRUM LOCA (FSLOCA) evaluation model (EM) for Turkey Point.

The NRC-approved FSLOCA methodology is described in the NRC-approved TR WCAP-16996-NP-A, Revision 1, “Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)” (Reference 26). The issuance of amendment for implementation of FSLOCA at Turkey Point was provided by letter dated May 24, 2022 (Reference 27).

The FSLOCA methodology evaluates the full spectrum of LOCA breaks that result from a postulated break in the RCS. The break sizes covered by the methodology include any breaks in which the flow is beyond the capacity of the normal charging pumps, up to and including a double-ended guillotine (DEG) rupture of an RCS cold leg with a break flow area equal to two times the pipe area. These break sizes include break spectrum defined as small and large break LOCAs.

In the methodology, the break size spectrum is divided into two regions. Region I provides coverage of cold leg breaks with an inventory loss just exceeding the capability of the normal charging pumps to **[[**

]]. The breaks in the Region I analysis do not include breaks smaller than 1-inch diameter, since those break sizes are non-limiting. Region II analysis simulations include breaks above 1.0 ft^2 break area, up to a maximum size of a DEG break. The FSLOCA methodology explicitly considers the effects of fuel pellet thermal conductivity degradation (TCD) and other burnup-related effects.

3.4.3.2 LOCA ECCS Analysis

The licensee performed FSLOCA evaluation for a range of operating conditions, which are provided in table 4.8-2 of the LAR. The table also includes several key parameters used in the analysis. In response to NRC staff questions, by letter dated October 3, 2024 (Reference 3), the licensee provided the FSLOCA evaluation report for Turkey Point. The report provides details on the FSLOCA EM development, computer codes used in the evaluation model, and input parameters and assumption used in the analysis. The report also includes the pertinent results for Region I and Region II analysis. The licensee stated in its response that the input parameters that correspond to the TS were identified at the beginning of the analysis process and appropriate uncertainty is accounted for in the analysis. A comparison of plant operating parameters with its TS limit was provided in table 14a of the licensee response, dated October 3, 2024. In the response, the licensee also stated that no reductions in conservatism were made for this analysis compared to the previous application of the FSLOCA methodology.

In response to the NRC staff question regarding difference in form loss coefficient between co-resident fuel assemblies in a mixed core, the licensee responded by letter dated October 3, 2024, that **[[**

]]. The licensee further stated that **[[**

]].

Based on its evaluation of the licensee provided justification, the NRC staff agrees that a transition core evaluation is not needed for FSLOCA for the proposed fuel transition.

3.4.3.3 Acceptance Criteria

Consistent to the 10 CFR 50.46(b) acceptance criteria, the licensee used the following ECCS acceptance criteria for the Turkey Point 24-month fuel cycle analysis with AXIOM cladding, ADOPT fuel pellets, and the PRIME fuel design:

- Peak Cladding Temperature (PCT) ≤ 2,200°F
- Minimum Equivalent Clad Reacted (ECR) Margin (MEM) ≥ 0 percent
- Maximum Core Wide Oxidation (CWO) ≤ 1 percent
- Core Coolable Geometry Maintained

The NRC staff notes that the MLO criterion of 17 percent from 10 CFR 50.46(b) is **[[ ]]**. The use of **[[ ]]** for AXIOM cladding is discussed in TR WCAP-18546-NP-A, where the **[[ ]]** is confirmed to remain below the ductile-to-brittle transition for AXIOM cladding.

3.4.3.4 Results

The evaluation performed to support the 24-month cycle fuel transition showed that the acceptance criteria are met for the Region I and Region II evaluations. **[[ ]]**.

The PCT evaluation, which corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level, shows that the PCT for the break spectrum analyzed does not exceed the 2,200°F limit.

The MEM evaluation, which corresponds to a bounding estimate of the 95th percentile MEM at the 95-percent confidence level, shows that the MEM is ≥ 0 percent for the break spectrum analyzed and is below the ductile-to-brittle transition.

The CWO evaluation, which corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level, shows that the CWO is ≤ 1 percent for the break spectrum analyzed.

Compliance with the maintaining coolable geometry criterion is met by demonstrating compliance with the PCT, MEM, and CWO criteria, and by assuring that fuel assembly grid deformation due to combined LOCA and seismic loads is specifically addressed. TR WCAP-16996-NP-A states that the effects of LOCA and seismic loads on the core geometry do not need to be considered unless fuel assembly grid deformation extends beyond the core periphery (i.e., deformation in a fuel assembly with no sides adjacent to the core baffle plates). The licensee evaluation showed that inboard grid deformation due to combined LOCA and seismic loads is not calculated to occur for Turkey Point.

The results of the FSLOCA evaluation for the 24-month cycle fuel transition are presented in table 4.8-1 of the LAR and are reproduced in table 2 below:

<u>Table 2: Turkey Point FSLOCA Results Summary</u>

Outcome	Region I Analysis Value	Region II Analysis Value (OPA)	Region II Analysis Value (LOOP)	Criterion
95/95 PCT	1495°F	1986°F	2047°F	≤ 2200°F
95/95 MEM	5.18%	2.18%	1.51%	≥ 0%
95/95 CWO	0.07%	0.83%	0.95%	≤ 1%

3.4.3.5 Compliance with NRC Imposed Limitations and Conditions

The NRC staff evaluation of compliance with the L&Cs listed in the SEs of the TRs used in the LOCA ECCS analysis is given below. This evaluation is based on the staff's review of Reference 1, Enclosure 2.

WCAP-16996-NP-A

L&C #1

> The FSLOCA™ EM applicability for performing PWR LOCA analyses is defined in terms of applicable accident transient phases so that the FSLOCA™ EM cannot be applied for analyzing the long-term core cooling phase of LOCA transients for the purpose of demonstrating compliance with the long-term core cooling requirement set forth in 10 CFR 50.46(b)(5). This limitation specifically addresses the condition that the FSLOCA™ EM does not treat boric acid precipitation and therefore lacks capabilities to address adequately post-LOCA long-term core cooling. The numerical approximations to advection and diffusion in the WCOBRA/TRAC-TF2 code conservation equations have neither been validated nor shown to successfully track the movement of high concentrations of boric acid between the vertical and radial cells with the vessel volumes.

Compliance with L&C #1

The licensee stated that the analysis for the Turkey Point 24-month cycle extension with fuel transition performed using the FSLOCA EM is not being used to demonstrate compliance with 10 CFR 50.46(b)(5) acceptance criteria.

L&C #2

> The FSLOCA™ EM applicability for performing PWR LOCA analyses is defined in terms of applicable types of PWR plants so that the EM can be applied for LOCA analyses of Westinghouse-designed three-loop and four-loop PWR plants with cold side emergency core cooling injection, only. Plant-specific applications will generally be considered acceptable if they follow the requirements pertinent to FSLOCA described in WCAP-16996- P/WCAP-16996-NP, Rev. 1, [(Reference 26)] and comply and meet the NRC limitations and conditions in this table (where the later document supersedes the earlier document when differences exist). Plant-specific licensing actions referencing FSLOCA analyses should include a statement summarizing the extent to which the FSLOCA methods and modeling were followed, and justification for any departures. Should NRC staff review determine that absolute adherence to the modeling guidelines is inappropriate for a specific plant, additional information may be requested using the RAI process.

Compliance with L&C #2

The licensee stated that Turkey Point, Unit Nos. 3 and 4 are Westinghouse-designed 3-loop PWRs with cold side injection, so they are within the NRC-approved methodology. The analysis for Turkey Point utilized the NRC-approved FSLOCA methodology with the exception of the changes that were previously transmitted to the NRC pursuant to 10 CFR 50.46. The licensee stated that the analysis was performed with a code version with the errors reported pursuant to 10 CFR 50.46, but that these errors were determined to have a negligible effect on the calculated results.

L&C #3

> The coupled WCOBRA/TRAC-TF2 and COCO codes or standalone LOTIC2 code will be applied to calculate the containment backpressure in PWR LOCA analyses for Region II so that a conservatively low, although not explicitly bounded, containment pressure will be predicted and used. For this purpose, the input to the COCO model and its prediction results will be based on appropriate plant-specific containment design parameters and initial conditions and will simulate accordingly engineered safety features and installed systems capable of affecting the containment pressure including their actuation, performance, and associated processes. The following specific limitations will apply for Region II analyses using the FSLOCA™ EM: (1) an acceptable plant-specific initial containment temperature will be determined based on input from the utility for the purpose of modeling the containment pressure response with COCO or LOTIC2; and (2) unqualified or indeterminate coatings throughout containment and qualified coatings within the break jet zone-of-influence will not be credited for the purpose of modeling the containment pressure response using COCO or LOTIC2 consistent with the bounding treatment of this parameter (conservatively low containment pressure)... .

Compliance with L&C #3

The licensee stated that the containment pressure calculation for the Turkey Point, Unit Nos. 3 and 4 analysis was performed consistent with the NRC-approved methodology. The licensee stated that appropriate design parameters and conditions were modeled, as were the engineered safety features that can reduce the containment pressure. A plant-specific initial temperature associated with normal full power operating conditions was modeled by the licensee, and no coatings were credited on any of the containment structures.

L&C #4

> As implemented by Westinghouse and found acceptable from the review of the decay heat model in the FSLOCA™ EM, the following conditions will apply with regard to decay heat modeling and sampling in PWR LOCA analyses for Region I and Region II: (1) decay heat uncertainty will be **[[ ]]** in uncertainty analyses for both Region I and Region II according to table 29-4 in WCAP-16996-P/WCAP-16996-NP, Revision 1, Volume III, section 29 (Reference 26); (2) the FSLOCA™ EM cannot be applied for transient time longer than 10,000 seconds following shutdown unless the decay heat model is shown to be acceptable for the analyzed core conditions. The latter limitation is **[[ ]]**.
>
> The sampled value of the decay heat uncertainty multiplier, DECAY_HT, reported in units of σ and absolute units, as applied for the limiting runs in Region

> I and Region II in the plant-specific analysis as part of an LAR submittal, will be provided as part of the submittal.

Compliance with L&C #4

The licensee stated that the decay heat uncertainty multiplier was **[[ ]]** consistent with the NRC-approved methodology. All of the analysis simulations were executed for no longer than 10,000 seconds following reactor trip and sampled values of the decay heat uncertainty multiplier for the cases that produced the Region I and Region II analysis PCT, MEM, and CWO results were provided by the licensee in units of sigma and approximate absolute units.

L&C #5

> The maximum assembly average burnup will be limited to **[[ ]]** and the maximum peak rod length-average burnup will be limited to **[[ ]]** within the FSLOCA™ EM. See Reference 26, WCAP-16996-P, Revision 1, section 32.4, Methodology Limitations, page 32-21.

Compliance with L&C #5

The licensee stated that the maximum analyzed assembly and rod length-average burnup for Turkey Point were less than or equal to **[[ ]]**, respectively.

L&C #6

> In the FSLOCA™ EM applications for PWR LOCA analyses, the latest version of an NRC-approved version of the latest fuel performance code that is applicable for the LOCA analysis will be used to initialize the fuel rod initial conditions. If the PAD 5.0 code is the latest approved version for fuel performance LOCA evaluations, then this version will be used to interface with WCOBRA/TRAC-TF2. The fuel performance code utilized shall be used to initialize WCOBRA/TRAC-TF2 using appropriate calculative methods to maximize the initial fuel stored energy and gap pin pressure, as well as adhere to any restrictions and limitations that resulted from the staff review and acceptance. The fuel performance code calculative methods should therefore exercise those modeling techniques approved by the staff for initializing WCOBRA/TRAC-TF2 for LOCA evaluations. The fuel performance code shall also include the effects of fuel TCD and its attendant effects on fuel rod behavior for application to the WCOBRA/TRAC-TF2 code.

Compliance with L&C #6

The licensee stated that PAD5 fuel performance data were utilized in the Turkey Point analysis with the FSLOCA EM. The licensee noted that PAD5 is licensed to model ADOPT fuel pellets per WCAP-18482-NP-A.

L&C #7

As implemented by Westinghouse and found appropriate based on the review of the two phase interfacial drag model of the 3D VESSEL module in WCOBRA/TRAC-TF2 and its assessment, the interfacial drag multiplier, YDRAG, applied to the small bubble, small-to-large bubble, and churn-turbulent flow regimes of the “Cold Wall” two-phase flow map and to the “Hot Wall” two-phase flow map interfacial drag will be **[[**

]] established for YDRAG in the FSLOCA™ EM as described in WCAP-16996-P/WCAP-16996-NP, Revision 1, section 13.4 and section 29.1.5 as lower interfacial drag reduces the two-phase mixture thus promoting core uncovery. This **[[**

]]. The comprehensive list of **[[**
]] is given in Table 29.2.3-1 of WCAP-16996-P, Revision 1 (Reference 26, see page 29-52).

Compliance with L&C #7

The licensee stated that the YDRAG uncertainty parameter was **[[**

]] for the Turkey Point Region I FSLOCA analysis.

L&C #8

As implemented by Westinghouse and found acceptable from the review of the corresponding WCOBRA/TRAC-TF2 models, certain uncertainty contributors will be **[[** **]]** for Region I analyses with the FSLOCA™ EM according to table 29.2.3-1 and table 29-2 in WCAP-16996-P/WCAP-16996-NP, Revision 1, Volume III, section 29.2.3. Specifically, the **[[**

]] as established in the FSLOCA™ EM and described in WCAP-16996-P, Revision 1, section 17.2.3 and section 29.1.6 for KCOSI and in section 4.4.5 and section 29.1.7 for HS_SLUG. Lower condensation heat transfer in the cold legs may influence depressurization rate during an SBLOCA boil-off period. A higher transition boundary delays transition to non-stratified flow thus increasing residual liquid in the loop seal regions and decreasing vapor venting capacity. These **[[**

]].

> **[[**
>
> **]]** can be found in tables 29-1, 29-2, 29-3a, 29-3b, 29-4, and 29-5 in WCAP-16996-P, Revision 1 (Reference 26, see pages 29-5 through 29-11). A compilation of the uncertainty parameter values and ranges can also be found in table I of LTR-NRC-15-85.
>
> Also note that with either of these above references, **[[**
>
> **]]** as documented in LTR-NRC-15-102, Revision 2 [(See Reference 26)].

<u>Compliance with L&C #8</u>

The licensee stated that consistent with the NRC-approved methodology, the **[[**

]] for the Turkey Point Region I FSLOCA analysis.

<u>L&C #9</u>

> In PWR plant type-specific applications of the FSLOCA™ EM for designs which are not Westinghouse 3-loop PWRs, a confirmatory evaluation will be performed for Region I analyses to assess the effect associated with the **[[**
>
> **]]**. This confirmatory evaluation will be performed once for each PWR plant type (e.g., Westinghouse design four-loop PWR plant) analyzed with the FSLOCA™ EM and referenced in subsequent plant-specific FSLOCA™ analyses of the same PWR plant type.

<u>Compliance with L&C #9</u>

The licensee stated that since the Turkey Point units are Westinghouse-designed 3-loop PWRs, this L&C is not applicable.

<u>L&C #10</u>

> In PWR plant type-specific application of the FSLOCA™ EM for designs which are not Westinghouse 3-loop PWRs, a confirmatory evaluation will be performed to demonstrate that the applied break size boundary between Region I and Region II serves the intended goal of **[[**
>
> **]]**. As of part this evaluation, it will be demonstrated that no unexplained behavior in the predicted safety criteria, including PCT, occurs across the boundary between Region I and Region II. In addition, it will be confirmed that **[[**

> **]]**. In addition, it is important to also assure that the limiting small break between about 2- and 4-inch in an equivalent break diameter is properly captured by the robust Region I analysis approach. Plants with larger RCS fluid volumes than the Beaver Valley plant test example in WCAP-16996-P/WCAP-16996- NP, Revision 1, should cover the same 2- to 4-inch range using break area to RCS volume scaling to assure that the 2- to 4-inch break range is preserved and not artificially truncated. This confirmatory evaluation will be performed once for each PWR plant type (e.g., Westinghouse design four-loop PWR plant) analyzed with the FSLOCA™ EM and referenced in subsequent plant-specific FSLOCA™ analyses of the same PWR plant type. The WCOBRA/TRAC-TF2 code is applicable for analysis over the entire break spectrum of LOCA transients. However, for the purpose of the Region II analysis, the minimum of the break area sampling should extend only to 1.0 ft^2 consistent with the ASTRUM LBLOCA EM (WCAP-16009-P-A, "Realistic Large-break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," Revision 0) in lieu of the Region I/II boundary.

Compliance with L&C #10

The licensee stated that since the Turkey Point units are Westinghouse-designed 3-loop PWRs, this L&C is not applicable.

L&C #11

> For each analysis performed using the FULL SPECTRUMTM LOCA methodology, the **[[ ]]**, seed, analysis inputs, and **[[ ]]** to be used for the Region I and Region II uncertainty analyses will be declared and documented prior to performing the uncertainty analyses. The **[[ ]]** will not be adjusted as a result of the outcome. Should a plant-specific application of the FSLOCA™ EM deviate from the originally declared analysis inputs for the intended purpose of demonstrating compliance with the applicable acceptance criteria, all modification(s) will be discussed in a calculation file and in the ECCS analysis submittal to NRC, as applicable, to explain the applicable reasons for the modification(s). In this instance, the analysis inputs will be modified only for the purpose of reflecting the implemented and described modeling changes. In addition, the calculated preliminary values for PCT, MLO, and CWO for each such case will be summarized for information only in the ECCS analysis submittal to the NRC. Because these preliminary analyses and results are not intended to demonstrate compliance with the criteria of 10 CFR 50.46, formal Appendix B verification and archival documentation of the underlying analyses are not required. Furthermore, operating ranges used in a plant-specific analysis as part of the sampling uncertainty analysis for Regions I and II are to be supplied for review by the NRC in a table format for both regions. In plant-specific reviews, the uncertainty treatment for such plant operating parameters including the sampled distributions and ranges will be considered acceptable if they meet or exceed corresponding design basis and/or TS LCO limits, with uncertainties included, as appropriate

(note: this condition should not be construed to imply that exceeding limiting values by any amount is acceptable; sampling distributions for plant parameters should be realistic and justifiable). Alternative approaches may be used, provided they are supported with appropriate justification. **[[**

]] are given in table 1 of LTR-NRC-17-47. Note that **[[** **]]** as per limitation no. 15 below.

<u>Compliance with L&C #11</u>

The licensee stated that for Turkey Point this L&C was met since:

- The **[[** **]]**, the Region I and Region II analysis seeds, and the analysis inputs were declared and documented prior to performing the Region I and Region II uncertainty analyses. The **[[** **]]**, and the Region I and Region II analysis seeds were not changed once they were declared and documented.

- The analysis inputs were not changed once they were declared and documented.

- The plant operating ranges sampled within the uncertainty analyses were provided for Turkey Point.

<u>L&C #12</u>

In plant-specific applications of the FSLOCA™ EM, a check will be performed to confirm that effects associated with dynamic pressure losses from the steam generator secondary side to the MSSVs are properly considered and adequately accounted for in the plant model used for the design basis LOCA analyses consistent with NRC Information Notice 97-09, “Inadequate Main Steam Safety Valve (MSSV) Set-Point and Performance Issues Associated with Long MSSV Inlet Piping.” SBLOCA performance is dependent on secondary pressure as it establishes primary pressure, and the consequential ECCS injection rate and potential for and degree of core uncovery.

<u>Compliance with L&C #12</u>

The licensee stated that bounding plant-specific dynamic pressure loss from the steam generator secondary side to the MSSVs was modeled in the Turkey Point analysis.

<u>L&C #13</u>

In plant-specific applications of the FSLOCA™ EM, **[[**

]] in the PWR model used to perform the design basis LOCA transient calculations, to capture the proper core two-phase

level response should the core uncover. Additionally, the **[[**
]] in such calculations. See Section 29.5.3, Venting, page 29-141 of WCAP-16996-P, Revision 1.

Compliance with L&C #13

The licensee stated that the analysis of Turkey Point models **[[**

]] in the analysis.

L&C #14

> For demonstration of compliance with the current 10 CFR 50.46 oxidation criterion, the oxidation result using Baker-Just to convert the LOCA transient time-at-temperature to an equivalent cladding reacted shall be compared against the 17 percent limit. If Cathcart- Pawel is used to convert the LOCA transient time-at-temperature to an equivalent cladding reacted, the oxidation result shall be compared to a 13 percent limit with the pre-transient oxide layer thickness being included in the prediction results. Should this measure (Cathcart-Pawel) 13 percent limitation not be carried forth to other NRC approvals of new realistic applications or should the value be changed, this SE and the two associated restrictions will be subsequently revised….

Compliance with L&C #14

The licensee stated that no changes to the current 10 CFR 50.46 acceptance criteria have been identified for AXIOM cladding. The licensee further stated that **[[**

]].

The licensee further stated that in compliance with the embrittlement criterion, the Cathcart-Pawel (CP) equivalent cladding reacted (ECR) will be confirmed to remain below the applicable ductile-to-brittle transition limit for consistency with the test data providing the basis for the ductile-to-brittle transition for AXIOM cladding described in section 3.11 of TR WCAP-18546-NP-A.

L&C #15

> Identification of the offsite power availability limiting condition for the Region II FSLOCA™ evaluation is required by GDC 35. In lieu of the method proposed by Westinghouse for addressing this requirement described in LTR-NRC-15-102, Revision 2, page 25, plant-specific applications of the FSLOCA™ EM should include two complete sets of sampled statistical evaluations: (1) a complete set with offsite power available, and (2) a second complete sampling set without offsite power available. For each set, the calculated statistical results at the 95/95 probability, confidence level should be demonstrated to comply with regulatory limits for PCT, MLO, and CWO. The **[[**
> **]]** to provide the required 95/95 probability, confidence statement that addresses the three major criteria of PCT, MLO, and CWO. This condition should be consistent with limitation number 11 in the table for **[[**
> **]]** for each sample set.

Compliance with L&C #15

The licensee stated that the **[[**
]]. Further, the Region II uncertainty analysis for Turkey Point was performed twice: once assuming LOOP, and once assuming OPA, and the results are in compliance with the ECCS acceptance criteria.

WCAP-18546-NP-A

Compliance with the limitations and conditions of the TR WCAP-18546-NP-A is discussed in section 3.4.1 of this SE.

Based on its review of the LAR, the NRC staff finds that all the applicable limitations and conditions from the TRs listed in the LOCA ECCS analysis have been met.

3.4.3.6 Conclusion

The NRC staff finds the Turkey Point FSLOCA break spectrum analysis to support the 24-month fuel transition to be acceptable based on the use of NRC-approved calculation methodology, compliance to the limitations and conditions from applicable topical reports, and meeting the acceptance criteria consistent with the 10 CFR 50.46(b) requirements.

3.4.4 Non-LOCA Events Analysis

Due to the proposed transition to a 24-month fuel cycle with new fuel and other changes listed below, the UFSAR Chapter 14 non-LOCA events are affected. The T-H performance and neutronics input changes could affect the safety analysis because they may potentially challenge the DNBR and FCM, specified acceptable fuel design limits, and other event-specific criteria such as time-to-criticality for the boron dilution event.

3.4.4.1 Analysis

The licensee stated that both minimum and maximum fuel temperatures are higher for ADOPT fuel than the existing temperatures for UO_2 fuel. Thus, the previous evaluation of minimum fuel temperatures for the implementation of PADS remains valid for affected events that were not reanalyzed for 24-month fuel cycles. The events that model maximum fuel temperatures were all reanalyzed for 24-month fuel cycles.

Proposed Changes in the Analysis Parameters

The licensee's proposed changes related to the 24-month fuel cycle with new fuel that affect inputs to the safety analyses of the non-LOCA events are summarized under headings (a) through (j) below.

(a) PRIME advanced fuel features

The licensee description of the PRIME fuel features consists of a reinforced dashpot, Low Tin ZIRLO mid-grids, and a low pressure drop bottom nozzle. The licensee stated that among these features, the pressure drop associated with the bottom nozzle affects the non-LOCA events analysis. The change results in a reduction of about 1 psi in the total core pressure drop, and a reduction in the core inlet pressure loss coefficient modeled in the WCAP-7907-P-A (Reference 28) methodology based on LOFTRAN code, and WCAP-14882-P-A (Reference 29) and WCAP-14882-S1-P-A (Reference 30) methodology based on RETRAN-02 code models used to reanalyze the events as applicable.

(b) AXIOM advanced fuel rod cladding material

The existing zirconium-based (Optimized ZIRLO) fuel cladding is replaced by AXIOM fuel cladding which has improved corrosion resistance, lower hydrogen pickup, and lower creep.

(c) ADOPT fuel pellets

The existing standard UO_2 fuel pellets are replaced by ADOPT fuel pellets with chromia and alumina doping which has increased uranium density and enables higher burnup and improved accident tolerance.

(d) Integral Fuel Burnable Fuel Absorbers (IFBA) and gadolinia burnable absorbers

The first use of gadolinia will be introduced in the currently used IFBA (zirconium diboride burnable absorber) which will affect the fuel properties modeled in the non-LOCA analysis codes.

(e) Increase maximum SG tube plugging (SGTP)

SGTP level is increased from 10 percent to 15 percent.

(f) Doppler-only power defect

The BOC least-negative Doppler-only power defect modeled in RCCA withdrawal from subcritical (RWFS) and rod cluster control assembly ejection analysis is changed from -1.19 percent $\Delta\rho$ to -1.11 percent $\Delta\rho$.

(g) Shutdown margin (SDM)

The SDM is decreased from 1.77 percent $\Delta\rho$ to 1.7 percent $\Delta\rho$

(h) MTC for ATWS

The most positive hot full power (HFP) MTC modeled for the ATWS event is increased from -8 pcm/°F to -7.6 pcm/°F.

(i) Trip reactivity curve

The revised trip reactivity curve inserts more negative reactivity than the current curve at a given rod position in the earlier portion of rod insertion, but slightly less reactivity in the later portion of the curve for rod position greater than 60 percent of the insertion. Both curves insert the full 4 percent $\Delta\rho$ trip reactivity at full insertion. The reactor coolant partial loss of flow (PLOF), complete loss of flow (CLOF), and RCP locked rotor events reanalyzed for the fuel cycle transition with new fuel incorporated the revised trip reactivity curve.

(j) Protection system setpoints and initial conditions uncertainties

The licensee has revised some of the RPS SAL setpoints and initial condition uncertainties (ICUs) either to incorporate changes that have been made since some of the existing non-LOCA were performed or to support the fuel cycle transition with the new fuel. Attachment 4 of the LAR describes these changes, which are evaluated by the NRC staff in sections 3.2 and 3.3 of this SE.

<u>Analysis of Events</u>

The NRC staff evaluation of the licensee's reanalysis of the non-LOCA events based on the proposed changes is given under headings (a) through (r) below.

(a) Uncontrolled RCCA Withdrawal from a Subcritical or Low-Power Startup Condition (UFSAR, section 14.1.1)

The uncontrolled RCCA withdrawal from a subcritical reactor is classified as an ANS Condition II event of moderate frequency. This event is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of RCCA resulting in a power excursion. This transient can be caused by a malfunction of the reactor control rod drive system which could occur while the reactor is subcritical or at power.

Reference 3, Enclosure 1, Attachment 1, section 1.1 provides analysis, description and methods used, cases analyzed, inputs and assumptions, acceptance criteria, and results.
The licensee used NRC-approved WCAP-7979-P-A (Reference 31) methodology based on TWINKLE code, WCAP-7908-P-A (Reference 32) methodology based on FACTRAN code, and WCAP-14565-P-A (Reference 25) methodology based on VIPRE code to analyze the transient. The TWINKLE code calculates the core average nuclear power transient. The FACTRAN code uses the core average nuclear power calculated by TWINKLE code to perform a fuel rod transient heat transfer calculation and determines the core average heat flux and hot spot temperature transient. The VIPRE code performs the DNBR calculations using the core average heat flux calculated by FACTRAN.

Reference 3, Enclosure 1, Attachment 1, table 1.1-1 presents the analysis sequence of events and figures 1.1-1 through 1.1 present the result graphs. Table 3 below provides the acceptance criteria and the results.

<u>Table 3: Acceptance Criteria and Results</u>

Criterion	Result	Acceptable Limit
DNBR	> **[[ ]]**	**[[ ]]** minimum (Note 1)
DNBR	> **[[ ]]**	**[[ ]]** minimum (Note 2)
FCM (°F)	2500	4937 (Note 3)
Notes: (1) For ABB-NV correlation below the first mixing vane grid. (2) For WRB-1 correlation above the first mixing vane grid. (3) Based on 8 weight percent gadolinia and maximum burnup of 65 GW/MTU		

The NRC staff finds the licensee's analysis acceptable because by using NRC-approved methodologies and conservative inputs and assumptions, the calculated results meet the acceptable limits.

(b) Uncontrolled RCCA Withdrawal at Power (UFSAR, section 14.1.2)
This event is classified as an ANS Condition II event. Reference 3, Enclosure 1, Attachment 1, section 1.2 provides analysis, description and methods used, cases analyzed, inputs and assumptions, acceptance criteria, and results.

The licensee used RETRAN methodology for DNB analysis, and LOFTRAN methodology for RCS overpressure analysis at 100 percent, 80 percent, 60 percent, and 10 percent of nuclear steam supply system power of 2,652 MWt.

Reference 3, Enclosure 1, Attachment 1, table 1.2-1 presents the analysis sequence of events and figures 1.2-1 through 1.2-8 present the result graphs.

The acceptance criteria limits and results of the most limiting case are given in table 4 below.

<u>Table 4: Acceptance Criteria (Limits) and Results (Reference 3, Enclosure 1, Attachment 1, Table 1.2-2)</u>

Case	Description	Results	Limit
DNBR	100% RTP, minimum reactivity feedback, RIR of 1 pcm/sec	**[[ ]]**	**[[ ]]** minimum

Case	Description	Results	Limit
Peak Core Heat Flux (fraction of nominal)	100% RTP, minimum reactivity feedback, RIR of 34 pcm/sec	1.15	1.20 maximum
Peak RCS Pressure (psia)	9% RTP, minimum reactivity feedback, RIR of 23 pcm/sec, minimum fuel UA, maximum delayed neutron fraction	2,737 (Note 1)	2748.5 maximum
Peak MSS Pressure (psia)	10% RTP, minimum reactivity feedback, RIR of 10 pcm/sec	1,182	1208.5 maximum
Note 1: The licensee calculated the peak RCS pressure based on 9% RTP and RIR of 23 pcm/sec. The licensee will confirm the maximum RIR on a reload-specific basis for the 24-month fuel cycles with the new fuel.			

The NRC staff finds the licensee's analysis acceptable because by using NRC-approved methodologies and conservative inputs and assumptions, the licensee's calculated results satisfy the acceptable limits.

(c) RCCA Drop (UFSAR, section 14.1.4)

As stated in UFSAR, section 14.1.4, a dropped RCCA event is an ANS Condition II event that is assumed to be initiated by a single electrical or mechanical failure which causes any number and combination of RCCAs from the same group of a given bank to drop to the bottom of the core. The resulting negative reactivity insertion causes nuclear power to rapidly decrease. Since this is a Condition II event, it must be shown that the DNB design basis is met for the combination of power, hot channel factor, and other system conditions that exist following the dropped RCCA(s).

The licensee performed a generic analysis using LOFTRAN code by choosing inputs that conservatively bound for all plants within the plant type (i.e., 2-loop, 3-loop, or 4-loop) ensuring that the transient statepoints generated produce limiting DNBR calculations. The inputs encompassed the key plant changes described in section 3.4.2 of this SE. The analysis confirmed that the dropped rod transient statepoints are not impacted.

The NRC staff finds the licensee's evaluation, confirming that the current analysis remains valid, acceptable because the generic analysis used for evaluation (a) used NRC-approved LOFTRAN code, (b) is valid for Turkey Point because it is a 3-loop Westinghouse PWR, and (c) used inputs that encompass the changes listed in section 3.4.2 of this SE.

(d) Chemical and Volume Control System Malfunction (UFSAR, section 14.1.5)

The Chemical and Volume Control System (CVCS) Malfunction event is classified as an ANS Condition II event. The analysis of this event considers boron dilution of the RCS during refueling, cold shutdown, hot shutdown, hot standby, startup, and power operation. UFSAR, section 14.1.5, describes the dilution of the RCS during these plant conditions. The licensee stated that an increased maximum SGTP level of 15 percent is considered in the revised analysis and that none of the other related key plant changes impact the event except that the confirmation of the resulting limits for some cases is affected by the change in SDM from the current licensing basis value of 1.77 percent $\Delta\rho$ (UFSAR, section 14.1.5) to a new value of 1.7 percent $\Delta\rho$.

Reference 3, Enclosure 1, Attachment 1, section 1.4 provides description, inputs and assumptions, acceptance criteria, and results of the boron dilutions analysis for the refueling, cold shutdown, hot shutdown, hot standby, startup, and power operation cases. Reference 3, Enclosure 1, table 1.4-2 provides sequence of events for this event.

The acceptance criterion applied to the CVCS malfunction event is that adequate time is available for operator action prior to a complete loss of SDM. For Modes 1, 2, 3, 4, and 5 there must be 15 minutes from event initiation to a complete loss of SDM, and for Mode 6 there must be at least 30 minutes from event initiation until the SDM is lost. Table 5 shows the calculated times to a loss of SDM for Modes 1, 2, and 6 and the calculated limits on the minimum permissible ratio of the initial boron concentration to the critical boron concentration for Modes 3, 4, and 5, all of which are confirmed to be met for the proposed change. Besides the acceptance criteria listed in table 5 below, the acceptance criteria for the RCS and MSS pressures are that they should be maintained below 110 percent of their design pressures.

Table 5: Acceptance Criteria and Results Summary (Reference 3, Enclosure 1, Attachment 1, Table 1.4-1)

Case Analyzed	Minimum Required Time (minutes)	Calculated Time to Criticality (minutes)	Calculated Limit on Ratio of Initial to Critical Boron Concentration
Mode 1 – Automatic Rod Control (Note 1)	15	32.0 (Notes 3)	N/A
Mode 1 – Manual Rod Control (Note 1)	15	30.0 (Note 3)	N/A
Mode 2 (Note 1)	15	17.5 (Note 3)	N/A
Mode 3	15.0 (Note 4)	≥ 15.0	1.060 (Note 2)
Mode 4 with RCP	15.0 (Note 4)	≥ 15.0	1.051 (Note 2)
Mode 4 with RHR [residual heat removal]	15.0 (Note 4)	≥ 15.0	1.100 (Note 2)
Mode 5 with RCS Filled	15.0 (Note 4)	≥ 15.0	1.092 (Note 2)
Mode 5 with RCS Drained	15.0 (Note 4)	≥ 15.0	1.112 (Note 2)

Case Analyzed	Minimum Required Time (minutes)	Calculated Time to Criticality (minutes)	Calculated Limit on Ratio of Initial to Critical Boron Concentration
Mode 5 with RCS Partly Drained	15.0 (Note 4)	≥ 15.0	1.105 (Note 2)
Mode 6 with RCS Filled (Note 1)	30	33.3	N/A
Mode 6 with RCS Drained (Mid-loop) (Note 1)	30	31.2	N/A

Notes:

1. For Modes 1, 2, and 6, as a part of the reload process, on a cycle-specific basis the licensee is to confirm the initial and critical boron concentration inputs.
2. For Modes 3, 4, and 5, as a part of the reload process, on a cycle-specific basis, in order to ensure that there is at least 15 minutes from the start of dilution to a complete loss of SDM, the licensee is to confirm the calculated limit on the minimum permissible ratio of initial and critical boron concentration.
3. For Modes 1 and 2, the calculated times are from the time at which alarm setpoint is reached to the time of a complete loss of SDM.
4. For Modes 3, 4, 5, and 6, the available times are from the time of event initiation to the time of a complete loss of SDM.

The NRC staff finds the licensee's analysis of the CVCS malfunction event acceptable because the licensee followed the currently used analysis method using acceptable inputs and assumptions. The calculated results show that for the boron dilution in the RCS caused by the CVCS malfunction, all acceptance criteria are met, i.e., in Modes 1 through 5 there is at least 15 minutes available from the time of event initiation to a loss of SDM, and in Mode 6 there is more than 30 minutes available from the time of event initiation to a loss of SDM. The RCS and MSS pressures are maintained below their 110 percent design pressures.

(e) Startup of an Inactive Reactor Coolant Loop (UFSAR, section 14.1.6) – Removed from licensing basis

The current plant TS prohibits plant startup and power operation (Modes 1 and 2) with one or more RCP loops out of service. Therefore, this event was removed from the Turkey Point licensing basis. Therefore, no evaluation is required based on the proposed change.

(f) Excess Feedwater Flow at Full Power and Reduction in Feedwater Enthalpy (UFSAR, section 14.1.7)

This event is classified as ANS Condition II event. It can occur due to a full opening of the feedwater control valve caused by control valve malfunction or an erroneous operator action. The current analysis was performed at HFP and HZP. The licensee stated that the HFP case is not reanalyzed as it was determined that none of the plant changes would adversely affect its current analysis. At HZP or no-load condition, it will cause RCS temperature reduction and therefore a reactivity increase due to the negative MTC of reactivity. The mitigation of this event takes place by automatic closure of feedwater control and isolation valves, closure of all feedwater bypass valves, trip of feedwater pumps, and a turbine trip on high-high SG water level.

The licensee used NRC-approved RETRAN code and VIPRE code methodologies for the reanalysis of this event. The RETRAN code simulated the RCS, core neutronics, the pressurizer, PORVs, PSVs, pressurizer sprays and heaters, SGs, and MSSVs. This code also computes pertinent plant variables including pressure, temperature, and power levels. The VIPRE code used the transient statepoint input from RETRAN and determined if the DNBR remains above its minimum limit.

Reference 3, Enclosure 1, Attachment 1, section 1.6.2 lists key inputs and assumption for this event analysis at HZP.

The key acceptance criteria are: (a) minimum calculated DNBR remains above its SAL, so that the fuel integrity is maintained and (b) RCS pressure and MSS pressure remain below 110 percent of their respective design pressures.

The licensee stated that the analysis resulted in a transient that is less severe than the HZP SLB event analysis described in Reference 3, Enclosure 1, Attachment 1, section 1.15 and evaluated by the NRC staff in section 3.4.2(p) of this SE. The licensee stated that the HZP SLB event analysis has the same initial conditions and is conservatively analyzed as an ANS Condition II event.

The NRC staff finds the licensee’s evaluation acceptable because (a) the DNBR results are bounded by the HZP SLB event DNBR results, (b) the excess feedwater flow tends to cool the RCS, therefore, the event will not challenge the RCS and the MSS pressure limits, and (c) the adverse impact on the steam turbine and steam piping due to excessive moisture carryover resulting from SG overfill due to excess feedwater flow is prevented by automatic feedwater isolation from high SG water level trip.

(g) Excessive Load Increase Incident (UFSAR, section 14.1.8)

As stated in UFSAR, section 14.1.8, an excessive load increase incident is defined as a rapid increase in SG steam flow causing a power mismatch between the reactor core power and the SG load demand. The RPS will trip the reactor for any loading rate increase beyond the capability of the reactor control system. In this case, the resulting transient is terminated in sufficient time to prevent DNBR from going below its limiting minimum value. The licensee stated that none of the proposed plant changes listed in section 3.4.2 of this SE adversely affect the current analysis and, therefore, the event is not reanalyzed for the proposed changes.
The NRC staff finds the licensee’s evaluation acceptable because the changes in inputs listed in section 3.4.2 of this SE do not impact the current analysis results.

(h) Loss of Reactor Coolant Flow (UFSAR, section 14.1.9)

The reactor coolant PLOF and CLOF can be caused by a mechanical or an electrical failure in one or more of RCPs or from a fault in the electrical power supply (such as under-voltage (UV) or under-frequency (UF)) to these pumps. A PLOF is considered to be an ANS Condition II event, an incident of moderate frequency. A CLOF-UV or CLOF-UF is considered to be an ANS Condition III event, an infrequent incident. The licensee stated that since a Condition II LOOP event can cause a complete loss of flow, analyses of all the loss of flow events are shown to meet Condition II acceptance criteria. The specific acceptance criteria for the loss of flow events are as follows.

- Fuel cladding integrity shall be maintained. This criterion is met by demonstrating that the minimum DNBR remains greater than the SAL value during the transient.

- Pressures in the RCS and MSS should be maintained below 110 percent of their respective design pressures.

The licensee stated that evaluation of the maximum RCS and MSS pressures for the loss of reactor coolant flow events is not necessary because it is bounded by the loss of load/turbine trip (LOL/TT) event evaluated in section 3.4.2(j) below. The LOL/TT transients are more severe for the resultant RCS and MSS pressures because the turbine trip is the initiating fault which trips the reactor, whereas for loss of flow events, the turbine trip occurs following reactor trip. The LOL/TT event analysis conservatively maximizes the calculated RCS and MSS pressures.

The licensee used the NRC-approved RTDP methodology (Reference 25). The computer codes used for reanalysis are NRC-approved RETRAN and VIPRE. The RETRAN code calculates the loop and core flow transients, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE code is then used to calculate the heat flux and DNBR transients based on the nuclear power and flow obtained from the RETRAN code output.

The reanalysis used nominal full core power, reactor vessel average temperature, pressurizer pressure, and minimum measured reactor coolant flow as initial conditions. The analysis incorporated the revised trip reactivity curve and updated minimum fuel temperatures. The analysis also incorporated the increased maximum SGTP level discussed in section 3.4.2 above. The licensee stated that SGTP level would have a negligible effect on the results because the limiting conditions for these events would occur much earlier than the time the SG characteristics can influence the results. None of the other key plant changes stated in section 3.4.2 above adversely impacted the analysis.

Reference 3, Enclosure 1, Attachment 1, section 1.8.1 provides description, inputs and assumptions, acceptance criteria, and results of the events, table 1.8.1-2 presents the analysis time sequence of events, and figures 1.8.1-1 through 1.8.1-12 present the result graphs.
The acceptance criteria limit for DNBR and results of the CLOF-UV, CLOF-UF, and PLOF cases are given in table 6 below.

Table 6: Acceptance Criteria and Results Summary for DNBR (Reference 3, Enclosure 1, Attachment 1, Table 1.8.1-1)

Analysis Case	Minimum DNBR (typical/ thimble)	Safety Analysis Limit DNBR
CLOF-UV	**[[ ]]**	**[[ ]]** Minimum
CLOF-UF	**[[ ]]**	
PLOF	**[[ ]]**	

The NRC staff finds the loss of reactor coolant flow analysis acceptable because (a) it is based on NRC-accepted RTDP methodology, and considering statistically the uncertainties in the initial conditions in defining the DNBR limit value as per this methodology, (b) it used NRC-approved RETRAN and VIPRE computer codes, (c) all other initial conditions were set to nominal values, (d) no credit was taken in the DNBR analysis for the increase in RCS pressure, (e) consistent with the current licensing basis analysis, no ESFs (e.g., SI system) were required to function, (f) no single active failure in any system or component required for mitigation of this event adversely affect the consequences of this event, and (g) the DNBR results meet the acceptance criteria.

(i) Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break (UFSAR, section 14.1.9)

An RCP rotor seizure and RCP shaft break, referred to as a locked rotor accident in UFSAR, Section 14.1.9, is considered to be an ANS Condition IV event because it is a limiting fault not expected to occur. It results in a rapid reduction in forced reactor coolant loop flow in the faulted loop and through the core, which results in a reactor trip on a low reactor coolant flow signal. The consequences of a locked rotor (i.e., an instantaneous seizure of a pump shaft) are very similar to those of a pump shaft break. The initial rate of the reduction in coolant flow is slightly greater for the locked rotor event. However, with a broken shaft, the impeller could conceivably be free to spin in the reverse direction. The effect of reverse spinning is to decrease the steady-state core flow when compared to the locked rotor scenario. This event increases the reactor coolant temperature and subsequently causes the fuel cladding temperature and RCS pressure to increase. If the reactor is at power at the time of this event, the immediate effect of the instantaneous reduction in the reactor coolant flow is a rapid increase in the reactor coolant temperature. If the reactor is not tripped promptly, the increase in the reactor coolant temperature could result in DNB with subsequent fuel damage.

The licensee used RTDP methodology in which the computer codes used are RETRAN and VIPRE. The RETRAN code calculates the loop and core flow transients, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE code is then used to calculate the heat flux and DNBR transients based on the nuclear power and flow obtained from the RETRAN code. The VIPRE code also performed the hot rod calculations to determine the fuel cladding temperature and amount of zirconium-water reaction versus time.

The licensee's analysis represents the most limiting condition for the locked rotor and pump shaft break accidents. The analysis simulated the event by modeling an immediate stop in the rotational speed of one RCP. This analysis is performed for case (a) and case (b) described below.

Case (a) maximizes the calculated percentage of rods-in-DNB:

The rods-in-DNB case analysis is used to determine the percentage of fuel failure, if any, for consideration in the locked rotor radiological analysis. A rods-in-DNB limit of 0 percent is used for consistency with the basis of the latest locked rotor radiological dose analysis that assumes no fuel failures. The acceptance criterion is that the total percentage of rods-in-DNB is less than that used in the radiological dose analysis.

Case (b) maximizes the calculated RCS pressure, fuel cladding temperature, and amount of zirconium-water reaction:

For calculating the peak RCS pressure and PCT, the licensee used nominal values of initial conditions for core power, reactor vessel average temperature, pressurizer pressure, pressurizer water level, and reactor coolant flow and included conservative uncertainty allowances and performed conservative analysis by not crediting the pressure-reducing effects of the pressurizer PORVs and sprays. The analysis modeled increased maximum fuel temperatures, minimum moderator temperature reactivity feedback, maximum Doppler reactivity feedback, maximum value for the delayed neutron fraction, and maximum 15 percent SGTP level. The analysis assumed control rods initially to be at their fully withdrawn position and conservatively used low trip reactivity value of 4.0 percent $\Delta\rho$ to minimize the effect of rod insertion following reactor trip.

The licensee also performed a separate calculation of the hot rod fuel cladding temperature using VIPRE code assuming that the rod is experiencing DNB throughout the transient and the rod power at the hot spot is 2.49 times the average rod power at the initial core power level using conservative inputs and assumptions.

Reference 3, Enclosure 1, Attachment 1, section 1.8.2 provides description, inputs and assumptions, acceptance criteria, and results of the analysis. Table 1.8.2-2 presents the analysis time sequence of events, and figures 1.8.2-1 through 1.8.2-3 present the result graphs.

Table 7: Acceptance Criteria and Results Summary for Locked Rotor Analysis (Reference 3, Enclosure 1, Attachment 1, Table 1.8.2-1)

Criterion	Result	Limit
Maximum Fuel Cladding Average Temperature at the Core Hot Spot, (°F)	1580.1	2375 (maximum)
Maximum Zirconium-Water Reaction at the Core Hot Spot, % by weight	0.16	16 (maximum)
Maximum RCS Pressure, (psia)	2710	2748.5

Criterion	Result	Limit
		(maximum) (Note 1)
Notes: 1. The licensing basis RCS pressure acceptance criterion is that pressure must not exceed that which would cause stresses to exceed the faulted condition stress limits of the RCS components. However, for conservatism, a more restrictive criterion of 110% of RCS design pressure is used.		

The NRC staff finds the licensee's analysis acceptable because of the following:

- The initial conditions are set to nominal values and uncertainties in the initial conditions in defining the DNBR limit are statistically determined following the NRC-approved RTDP methodology.
- The analysis is based on NRC-approved RETRAN and VIPRE computer codes.
- Conservatively, no credit is taken in the increase in RCS pressure for the DNBR analysis.
- Consistent with the current licensing basis analysis, no ESFs (e.g., SI system) are credited.
- No single active failure in any system or component required for mitigation of this event adversely affect the consequences of this event.
- The results of the locked rotor rods-in-DNB case analysis show that the minimum DNBR remains above the limit value, and there are no rods-in-DNB predicted.
- The results showed significantly less limiting fuel cladding temperature and percentage of zirconium reacted results, and a slightly more limiting peak RCS pressure result.
- Acceptance criteria of rods-in-DNB limit of 0 percent, maximum fuel cladding temperature at the core hot spot limit of 2375°F, maximum percentage of zirconium reacted at the core hot spot limit of 16 percent, and maximum RCS pressure limit of 2748.5 psia are met.

(j) Loss of External Electrical Load (UFSAR, section 14.1.10)

As stated in UFSAR, section 14.1.10, the loss of external electrical load may result from an abnormal decrease in network frequency or an accidental opening of the main breaker from the generator that fails to cause a turbine trip, but causes a rapid large load reduction by the action of the turbine control. For either case, offsite power is available for the continued operation of plant components such as the RCPs. A major loss of plant load can also result from a turbine trip. Since this event occurrence is considered to be of moderate frequency, it is classified as an ANS Condition II event.

For analyzing this event, referred to as the LOL/TT event, the licensee incorporated the increased maximum fuel temperatures and increased maximum SGTP level changes described in section 3.4.2 above. The licensee stated that these changes directly impact the analysis results while the other changes described in section 3.4.2 do not adversely impact the analysis of this event.

The licensee used NRC-approved RETRAN code to determine the plant transient conditions following an LOL/TT event. The code models the core neutron kinetics, RCS, pressurizer, pressurizer PORVs and sprays, SGs, MSSVs, and the AFWS. The code also computes pertinent variables, including the pressurizer pressure, pressurizer water level, SG mass, and reactor coolant average temperature.

The licensee analyzed the following three cases at HFP condition:

1. Minimum DNBR case with automatic pressurizer pressure control and maximum SGTP.

2. Peak MSS pressure case with automatic pressurizer pressure control and minimum SGTP.

3. Peak RCS pressure case without automatic pressurizer pressure control and maximum SGTP.

Reference 3, Enclosure 1, Attachment 1, sections 1.9.2.1 through 1.9.2.8 identify the plant initial operating conditions used in the analysis, section 1.9.1 describes the analysis, and section 1.9.4 provides description of the above three cases. The acceptance criteria are as follows:

- RCS and MSS pressures should remain below 110 percent of their respective design pressures (RCS pressure limit of 2,748.5 psia and an MSS pressure limit of 1,208.5 psia).

- Fuel cladding integrity shall be maintained by demonstrating that the minimum DNBR remains above the 95/95 DNBR limit (the applicable safety analysis DNBR limit is **[[ ]]**.)

- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. This criterion is satisfied by verifying that the pressurizer does not become water-solid (i.e., total pressurizer water volume remains less than 1,300 ft^3).

- An incident of moderate frequency in combination with any single active component failure, or single operator error, is considered an event for which an estimate of the number of potential fuel failures is provided for radiological dose calculations. For such accidents, fuel failure is assumed for all rods for which the DNBR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model, that fewer failures occur. There is no loss of function of any fission product barrier other than the fuel cladding. This criterion is satisfied by verifying that the minimum DNBR remains above the 95/95 DNBR limit.

Table 8: Loss of External Electrical Load/Turbine Trip Results Summary (Reference 3, Enclosure 1, Attachment 1, Table 1.9-2)

Parameter	**Result**	**Limit**
Minimum DNBR	**[[ ]]**	**[[ ]]** minimum

Parameter	Result	Limit
Peak RCS Pressure (psia)	2,747.87	2,748.5 maximum
Peak MSS Pressure (psia)	1,198.93	1,208.5 maximum

The NRC staff finds the analysis acceptable because the licensee used NRC-approved RETRAN code while changing the inputs specified in section 3.4.2 that impact the analysis. As shown in table 7, the DNBR and peak RCS and MSS pressures meet the acceptance criteria.

(k) Loss of Normal Feedwater Flow (UFSAR, section 14.1.11)

The loss of normal feedwater flow (LONF) is an event of moderate frequency and therefore is considered to be an ANS Condition II event. It can occur due to loss of AC power, pump failure, valve malfunction, or a complete loss of all AC power to plant auxiliaries. It results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core.

The licensee used NRC-approved RETRAN code to determine the plant transient conditions following an LONF event. The code models the core neutron kinetics, RCS, pressurizer, pressurizer PORVs and sprays, SGs, MSSVs, and the AFWS. The code also computes pertinent variables, including the pressurizer pressure, pressurizer water level, SG mass, and reactor coolant average temperature.

The licensee stated that the analysis incorporated the following changes listed in section 3.4.2 above: increased maximum fuel temperatures, increased decay heat, and increased maximum SGTP level. These changes directly impact the analysis results, while none of the other key changes adversely impact the analysis of this event. The licensee credited a portion of reactor coolant to metal heat transfer during the long-term primary side heatup using the thick metal mass heat transfer model in NRC-approved WCAP-14882-S1-P-A developed for use in the analysis of long-term RCS heatup events

The acceptance criteria associated with this event are as follows:

- Fuel cladding integrity shall be maintained by demonstrating that the minimum DNBR remains above the 95/95 DNBR limit.

- Pressures in the RCS and MSS should be maintained below 110 percent of their respective design pressures.

- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. During this event, the serious plant condition to be prevented caused by RCS heatup is for the pressurizer to become water-solid. This will preclude any water relief through the pressurizer PORVs or PSVs.

For both the DNBR and overpressurization criteria, the license compared the consequences of LONF with the LOL/TT event evaluated in section 3.2.9.10 above. Both events represent a reduction in the heat removal capability of the secondary system. However, the LOL/TT event is more severe than LONF for DNBR as well as for overpressurization because of the following reasons:

- The LONF event causes very little power mismatch between the primary and secondary systems because the RCS temperature increases gradually as the SGs boil down to the low-low water level trip setpoint, at which time reactor trip occurs, followed by turbine trip.

- The LOL/TT event initiates turbine trip followed by reactor trip which results in a much severe and longer period of power mismatch between the primary and secondary. As such, the initial RCS heatup will also be much more severe for the LOL/TT than for the LONF event.

Based on the above reasons, the LONF event DNBR and RCS and MSS overpressurization results will always be bounded by the LOL/TT event results. Therefore, the licensee did not explicitly analyze the LONF event with respect to the DNBR and RCS and MSS overpressurization. The LOL/TT event analysis described in section 3.2.9.10 demonstrates that the RCS and MSS maximum pressure limits of 2748.5 psia and 1208.5 psia, respectively, are met.

The licensee performed analysis to determine if the pressurizer does not become water-solid. This analysis will also confirm that the AFWS capacity is sufficient for long-term removal of the decay and RCP heat. The pressurizer water volume transient (figure 1.10-2 in Reference 3, Enclosure 1, Attachment 1) shows two distinct peaks. The first peak is a function of the initial conditions, while the second peak is an indication of the capability of the AFWS to provide long-term heat removal. Thus, the magnitude of the second peak is used to determine the limiting case.

Reference 3, Enclosure 1, Attachment 1, section 1.10.1 provides event description, section 1.10.2 provides input parameters and assumptions, section 1.10.3 provides analysis description and evaluation, and section 1.10.4 provides acceptance criteria and results of analysis. Table 1.10-1 presents the analysis time sequence of events, and figures 1.10-1 through 1.10-5 presents the result graphs.

Table 9: Acceptance Criteria and Results Summary for LONF Analysis (Reference 3, Enclosure 1, Attachment 1, Table 1.10-2)

Analyzed Case	Result	Limit to Prevent Pressurizer Water-Solid
Peak Pressurizer Water Volume for the Limiting Case (ft^3)	1,295	1,300 maximum

The results demonstrate that the pressurizer does not become water-solid in the long-term during the LONF event and, therefore, water relief through the pressurizer PORVs or PSVs will not occur. They also confirm that the AFWS capacity is sufficient for long-term sensible and decay heat removal and will prevent RCS or MSS overpressurization and core uncovery and satisfy the long-term DNBR limit criterion.

(I) Loss of Non-Emergency A-C Power to the Plant Auxiliaries (UFSAR, section 14.1.12)

A complete loss of non-emergency AC power (LOAC) may result in the loss of all power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite AC distribution system. Based on its expected frequency of occurrence, the LOAC event is considered to be an ANS Condition II event, an incident of moderate frequency.

The licensee used NRC-approved RETRAN code to determine the plant transient conditions following an LOAC event. The code models the core neutron kinetics, RCS, pressurizer, pressurizer PORVs and sprays, SGs, MSSVs, and the AFWS. The code also computes pertinent variables, including the pressurizer pressure, pressurizer water level, SG mass, and reactor coolant average temperature.

The licensee stated that analysis incorporated the following changes listed in section 3.4.2 above: increased maximum fuel temperatures, increased decay heat, and increased maximum SGTP level. These changes directly impact the analysis results, while none of the other key changes adversely impact the analysis of this event. The licensee credited a portion of reactor coolant to metal heat transfer during the long-term primary side heatup using the thick metal mass heat transfer model in NRC-approved WCAP-14882-S1-P-A (Reference 30) developed for use in the analysis of long-term RCS heatup events.

Consistent with the LONF analysis, in order to obtain additional margin to prevent pressurizer filling, the licensee modeled the end of cycle (EOC) maximum fuel temperatures and least-negative MTC with a revised SG low-low level safety analysis setpoint of 11 percent narrow range span.

The acceptance criteria associated with this event are as follows:

- Pressures in the RCS and MSS should be maintained below 110 percent of their respective design pressures.

- Fuel cladding integrity shall be maintained by demonstrating that the minimum DNBR remains above the 95/95 DNBR limit.

- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. During this event, the serious plant condition to be prevented caused by RCS heatup is for the pressurizer to become water-solid. This will preclude any water relief through the pressurizer PORVs or PSVs.

For the RCS and MSS overpressurization criteria, the license compared the consequences of LOAC with the LOL/TT event evaluated in section 3.2.9.10 above. Both events represent a reduction in the heat removal capability of the secondary system. However, the LOL/TT event is more severe than LOAC overpressurization because the LOL/TT event initiates turbine trip followed by reactor trip which results in a much more severe and longer period of power mismatch between the primary and secondary. As such, the initial RCS heatup will also be much more severe for the LOL/TT than for the LOAC event.

For the DNB criteria, the licensee stated that the LOAC event is bounded by the CLOF evaluated in section 3.4.2(h) above because of the reasons given below:

- In the CLOF event, the RCS flow coastdown is the initiating fault which trips the reactor after the flow is degraded.

- In the LOAC event, the flow coastdown occurs after reactor trip.

- The CLOF event will experience much less reactor coolant flow (occurs before reactor trip) than the LOAC event flow (occurs after reactor trip), therefore, for the CLOF event the DNB consequences will be more limiting.

- The CLOF event analysis described in section 3.4.2(h) demonstrates that the minimum DNBR remains above the SAL.

Based on above reasons, the LOAC event overpressurization is bounded by the LOL/TT overpressurization, and the LOAC event DNBR is bounded by the CLOF event DNBR. Therefore, the licensee did not explicitly analyze the LOAC event with respect to the DNBR and RCS and MSS overpressurization criteria. The LOL/TT event analysis demonstrates that the RCS and MSS maximum pressure limits of 2748.5 psia and 1208.5 psia respectively are met.

The licensee performed analysis to determine if the pressurizer does not become water-solid. This analysis will also confirm that the AFWS capacity is sufficient for long-term removal of decay and RCP heat. The pressurizer water volume transient (figure 1.11-2 in Reference 3, Enclosure 1, Attachment 1) shows two distinct peaks. The first peak is a function of the initial conditions, while the second peak is an indication of the capability of the AFWS to provide long-term heat removal. Thus, the magnitude of the second peak is used to determine the limiting case.

Reference 3, Enclosure 1, Attachment 1, section 1.11.1 provides event description, section 1.11.2 provides input parameters and assumptions, section 1.11.3 provides analysis description and evaluations, and section 1.11.4 provides the acceptance criteria and results of the analysis. Table 1.11-1 presents the analysis time sequence of events, and figures 1.10-1 through 1.11-5 present the result graphs.

<u>Table 10: Acceptance Criteria and Results Summary for LOAC Analysis (Reference 3, Enclosure 1, Attachment 1, Table 1.11.2)</u>

Analyzed Case	**Result**	**Limit to Prevent Pressurizer Water-Solid**
Peak Pressurizer Water Volume for the Limiting Case (ft^3)	1,224	1,300 maximum

The results demonstrate that the pressurizer does not become water-solid in the long-term during the LOAC event and, therefore, water relief through the pressurizer PORVs or PSVs will not occur. It also confirms that the AFWS capacity is sufficient for long-term sensible and decay heat removal and will prevent RCS or MSS overpressurization and core uncovery and satisfy the long-term DNBR limit criterion.

(m) Accidental Depressurization of the Reactor Coolant System (UFSAR, section 14.1 14)

An accidental depressurization of the RCS could occur from an inadvertent opening of a PORV, PSV, or pressurizer spray valve. Because of higher flow through a PSV, the depressurization caused by its inadvertent opening will cause much severe core conditions than resulting from a PORV or pressurizer spray valve inadvertent opening. The results of this analysis are shown to comply with the acceptance criteria for a Condition II event.

The licensee used NRC-approved RETRAN code to determine the plant transient conditions following this event. The code models the core neutron kinetics, RCS, pressurizer, pressurizer PORVs and sprays, SGs, MSSVs, and the AFWS. The code also computes pertinent variables, including the pressurizer pressure, pressurizer water level, SG mass, and reactor coolant average temperature.

The licensee used the same inputs and assumptions as in the current licensing basis analysis except for the increase in the SGTP level to 15 percent mentioned in section 3.4.2 above.

The acceptance criteria associated with this event are as follows:

- Pressures in the RCS and MSS should be maintained below 110 percent of their respective design pressures.

- Fuel cladding integrity shall be maintained by demonstrating that the minimum DNBR remains above the 95/95 DNBR limit.

- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

The RCS pressure limit is not challenged because it continuously decreases during the transient. The LOL/TT event evaluated in section 3.4.2(j) above is more severe than this event because turbine trip is its initiating fault, while in this event, turbine trip occurs following reactor trip. Therefore, the primary to secondary power mismatch and resultant RCS and MSS heatup and pressurization transients are more severe for the LOL/TT event.

The main acceptance criterion to be confirmed is that the minimum DNBR should remain greater than the SAL of **[[ ]]**. The licensee's analysis results for the minimum DNBR is **[[ ]]** which occurs at 39.5 seconds during the transient.

Reference 3, Enclosure 1, Attachment 1, section 1.12.1 provides event description, section 1.12.2 provides input parameters and assumptions, section 1.12.3 provides analysis description and evaluations, and section 1.12.4 provides the acceptance criteria and results of the analysis. Table 1.12-1 presents the analysis time sequence of events, and figures 1.12-1 and 1.12-2 presents the result graphs.

The NRC staff evaluation of this event is as follows:

- The NRC staff finds that the RCS and MSS overpressure is not required to be analyzed because it is bounded by the LOL/TT overpressure.

- The NRC staff finds the DNBR analysis and result acceptable because the licensee used NRC-approved RETRAN methodology while using the same inputs as in the current licensing basis with the exception of the revised and more conservative SGTP value of 15 percent. The calculated minimum DNBR **[[ ]]** is greater than the SAL **[[ ]]** and therefore meets the acceptance criteria.

(n) Anticipated Transients Without Scram (UFSAR, section 14.1.15)

As stated in UFSAR, section 14.1.5.1, an Anticipated Transient Without Scram (ATWS) event is defined as an Anticipated Operational Occurrence (AOO) followed by the failure of the reactor trip portion of the RPS. For Westinghouse PWRs, 10 CFR 50.62(c)(1) requires that each PWR must have equipment that is diverse from the reactor trip system to automatically initiate the AFWS and initiate a turbine trip under conditions indicative of an ATWS event.
The acceptance criterion for analyzing the ATWS event is to demonstrate that the peak RCS pressure does not exceed the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Service Level C stress limit criterion of 3200 psig. Consistent with the current licensing basis, the licensee analyzed this event with the NRC-approved LOFTRAN code. The licensee stated that none of the key plant changes discussed in section 3.4.2 above adversely affect the analysis with the exception of HFP MTC. The most-positive HFP MTC modeled in the current ATWS analysis is -8.0 pcm/°F. To assess the impact of MTC increase to -7.6 pcm/°F, the licensee analyzed the LONF and LOL/TT events under ATWS conditions. The licensee selected these two events because they result in maximum RCS overpressurization in the presence of the reactor trip system. As expected for both events, the peak RCS pressure increased but did not exceed the acceptance criterion of 3200 psig.

The NRC staff finds the ATWS analysis for the LONF and LOL/TT events acceptable because by using NRC-approved LOFTRAN code and increasing the MTC to -7.6 pcm/°F, the licensee determined that the RCS pressure does not exceed the maximum acceptable limit of 3200 psig.

(o) Inadvertent Opening of a Steam Generator Relief or Safety Valve (UFSAR, section 14.2.5.1)

This event occurs due to an accidental depressurization of the MSS caused by an inadvertent opening of any single secondary system valves, i.e., SG PORV, MSSV, or steam dump valve. UFSAR, section 14.2.5.1.1 states that since the effective steam flow area of these valves is less than a full double-ended rupture (DER) of a main steam line (or steam line break (SLB)), the RCS cooldown rate and resulting return to power are much less than the SLB. Therefore, with respect to the DNBR criterion, an accidental depressurization of MSS is always less-limiting than an SLB. The main concern in this and the SLB event is that minimum DNBR does not violate its SAL value. Based on its expected frequency of occurrence, an SLB event is considered to be an ANS Condition IV event. Evaluation of an SLB at full power is covered in section 3.4.2(p).

(p) Steam System Piping Failure at Full Power (UFSAR, section 14.2.5.2)

As described in UFSAR, section 14.2.5.2, the steam release from a main SLB would result in an initial increase in steam flow which decreases during the accident as the steam pressure decreases. The energy removal from the RCS causes a reduction of RCS temperature and pressure. The RCS cooldown results in an insertion of positive reactivity due its negative MTC. The current SLB analysis is performed at HFP and HZP.

The SLB at HFP is not reanalyzed because the licensee determined that none of the key plant changes discussed in section 3.4.2 adversely impact the current analysis. The licensee provided the following rationale (Reference 3):

> Only the revised trip reactivity curve discussed in [section 3.4.2] could potentially adversely impact the analysis of this event. The revised trip reactivity versus position curve provides the same or higher trip reactivity than the previous curve (used in the HFP SLB analysis), except for rod position points greater than 60 [percent] insertion and less than full insertion. Since the HFP SLB reaches its limiting transient condition during rod insertion (at 0.5 second after the rods begin to drop, per the analysis performed supporting PAD5 implementation), it is sensitive to the profile of the trip reactivity curve. Using the rod position versus time curve, this time was converted to rod position to determine the impact of the revised trip reactivity versus position curve on the event. Calculations determined that 0.5 second corresponds to a rod position of only about 6 [percent] rod insertion. Since the revised trip curve yields significantly more trip reactivity during the early part of rod insertion and the limiting time during the event occurs well before the revised trip curve becomes limiting (i.e., less reactivity inserted, at greater than 60 [percent] rod insertion), the results of the HFP SLB analysis are not adversely affected, and the analysis remains bounding and conservative.

The NRC staff agrees that the current HFP SLB analysis is not affected by the proposed changes based on the above rationale. Therefore, the current analysis in UFSAR remains valid.

For the SLB at HZP analysis, the licensee stated that none of the changes in section 3.4.2 impact the analysis except the reduced SDM acceptance criteria of 1.70 $\Delta\rho$. The licensee used NRC-approved RETRAN code to determine the plant transient conditions following the SLB event. The RETRAN code outputs core heat flux, RCS loop inlet temperatures, pressure, and core flow, which are used as inputs to the NRC-approved VIPRE code to determine if the DNBR limit is met.

Based on its expected frequency of occurrence, an SLB event is considered to be an ANS Condition IV event, a limiting fault. However, the licensee conservatively analyzed the event to Condition II acceptance criteria. The specific acceptance criteria applied in the analysis are as follows:

- The minimum DNBR during the transient should remain above the 95/95 DNBR limit. Based on WLOP correlation the minimum DNBR limit is **[[ ]]**.

- The peak linear heat generation rate (LHGR) in kW/ft should not exceed a value that would cause FCM. The peak LHGR for FCM is 22.7 kW/ft (Reference 3, Enclosure 1, Attachment 1, table 7.1-1), which is the same as the current design value.

- RCS and MSS pressures should be maintained below 110 percent of their respective design pressures.

Reference 3, Enclosure 1, Attachment 1, section 1.15.1 provides event description, section 1.15.2 provides input parameters and assumptions, section 1.15.3 provides analysis description and evaluation, and section 1.15.4 provides acceptance criteria and results of analysis. Table 1.15-1 presents the analysis time sequence of events, and figures 1.15-1 through 1.15-8 present the result graphs.

The most limiting main SLB at HZP is the case in which offsite power is assumed to be available, for which there is full reactor coolant flow throughout the transient. The calculated minimum DNBR is **[[ ]]** compared to the applicable minimum SAL value of **[[ ]]** based on WLOP DNBR correlation. The calculated LHGR did not exceed its limit. The RCS and MSS pressures remain below their design limit during the transient. The RCS and MSS pressures decrease from their initial values during the transient, and thus the pressure limits are not challenged for this event.

The NRC staff finds the SLB analysis at HZP acceptable because the licensee used NRC-approved computer codes and conservatively calculated the minimum DNBR during the transient to be greater than the minimum SAL. Additionally, the LHGR remains below its FCM limit and the RCS and MSS pressures are not challenged because the pressure continuously decreases during the transient. Based on this the NRC staff finds that the acceptance criteria are met.

(q) Rupture of a Control Rod Mechanism Housing – RCCA Ejection (UFSAR, section 14.2.6)

As described in UFSAR, section 14.2.6.1, only the initial few seconds of this transient are considered because the long-term considerations are the same as for LOCA. The licensee analyzed the RCCA ejection event for the proposed change using the 3-D rod ejection methodology described in NRC-approved TR WCAP-15806-P-A (Reference 36).

For the evaluation of RCS overpressure due to rod ejection, Reference 36 (Non-proprietary version), section 2.3, "RCS Overpressure Evaluation Method," states:

> An existing RCS overpressure evaluation of record will continue to be used if the core power transient from the 3-D evaluation is bounded by the transient from the reference case used in the existing overpressure evaluation.

The applicable RCS overpressure limit for the rod ejection event is 3200 psig (3215 psia), which corresponds to Level C Service Limits of the ASME B&PV Code. Confirmation that the RCS overpressure criterion is satisfied is generically addressed in WCAP-7588, Revision 1-A (Reference 33), section 4.4, in which the peak transient pressure for a limiting case was shown to not exceed 2800 psia. The NRC staff evaluation of 3-dimensional RCCA ejection analysis is addressed in section 3.4.12 of this SE.

(r) Feedwater System Pipe Break (UFSAR, section 14.2.7)

The feedwater line break (FWLB) event is considered to be a break in the feedwater pipe large enough to prevent the addition of sufficient feedwater to maintain the shell-side SG inventory. If the break occurs in the main feedwater line between the check valve and the SG, fluid from the

SG would be discharged through the break. Furthermore, because the AFW piping connects to the main feedwater line, a break between a main feedwater line check valve and the corresponding SG will prevent the addition of AFW to that SG. If a break occurs upstream of a feedwater line check valve, the transient will progress like a loss of normal feedwater event, where there is no loss of SG water inventory. Based on the break size and the plant operating conditions at the time the break occurs, this event could cause either a cooldown because of excessive energy discharge through the break, or a heatup of the RCS.

The licensee stated that the results of RCS overcooling resulting from a feedwater line break are bounded by the RCS overcooling results of an SLB because the higher enthalpy of steam translates into a greater heat transfer between the primary and secondary systems, and thus a greater RCS cooldown. Therefore, the licensee analyzed this event for RCS overheating. The feedwater flow reduction can cause the RCS temperatures to increase prior to reactor trip. Additionally, for the break located in the main feedwater line between the upstream check valve and the SG, the loss of SG fluid inventory through the break will reduce the heat sink volume available for decay heat removal following reactor trip.

Reference 3, Enclosure 1, Attachment1, section 1.17.1 provides a description of this event, section 1.17.2 provides inputs and assumptions, and section 1.17.3 provides analysis description and the licensee's evaluation. The licensee used NRC-approved RETRAN methodology for analysis and incorporated the increased maximum fuel temperatures, increased decay heat, and increased maximum SGTP level, which directly impact the analysis results. The analysis also credited the revised SG level ICU discussed in section 1.0.4.

The sequence of events is provided in Reference 3, table 1.17-1.

The specific acceptance criteria applied in the analysis are as follows:

- Core remains covered with water by demonstrating that there is no boiling in the RCS loops, thus confirming the adequacy of the AFW system for removing the sensible and decay heat.
- RCS and MSS pressures remain below 110 percent of their respective design pressures.
- Any activity release is such that the calculated doses are within design limits.

For the acceptance criterion of the core remaining covered, the licensee described the following two conditions:

- With respect to uncovering of the core and damage to the fuel cladding due to "dryout", the licensee stated that Westinghouse has established an internal criterion that no bulk boiling occurs prior to the event turnaround. The turnaround occurs when the heat removal capability of the SGs, fed by AFW, exceeds the sensible and decay heat generation in the RCS. This conservatively ensures that the core remains covered with water and thereby will remain in place and geometrically intact with no loss of core cooling capability. This criterion is conservative and is chosen for convenience in interpreting the FWLB event results.
- With respect to fuel cladding damage due to DNB, the licensee stated that the reactor pre-trip aspects of the FWLB event are bounded by the analysis of the loss of external electrical load event, while the reactor post-trip aspects of the FWLB event are bounded by the analysis of HZP SLB event.

The results of the licensee's analysis for two of several cases analyzed are as follows:

- Case 1 – for the break flow area 0.89 ft^2, the margin to the RCS saturation temperature before turnaround occurred is 13.9°F.

- Case 2 – for the break flow area 0.20 ft^2, the margin to the RCS saturation temperature before turnaround occurred is 2.7°F.

The results of the FWLB analysis showed that no bulk boiling occurred in the RCS prior to the time that the heat removal capability of the SGs being fed auxiliary feedwater exceeded the sensible and decay heat generation in the RCS. For the RCS and MSS pressures, the licensee stated that the analysis is bounded by the loss of external electrical load event in which the inputs are conservatively maximized for the calculation of RCS and MSS pressures. For the activity release, the licensee stated that no damage to the fuel cladding ensures that the dose will be within regulatory limits.

Reference 3, figures 1.17-1 through 1.17-5 show the results of the Case 1 analysis, and figures 1.17-6 through 1.17-10 show the results of the Case 2 analysis.

The NRC staff evaluation of the licensee's analysis of the FWLB event is as follows:

- The NRC staff finds it acceptable that the RCS overcooling that could be caused by the FWLB event is bounded by the RCS overcooling analysis of the SLB event because the higher enthalpy of steam from the SLB event translates into a greater heat transfer between the primary and secondary systems, resulting in a greater RCS cooldown.

- The NRC staff finds it acceptable that the adverse effect on the fuel cladding due to DNB from reactor pre-trip results of the FWLB event are bounded by the analysis of the loss of external electrical load event and the reactor post-trip results of the FWLB event are bounded by the analysis of HZP SLB event.

- The NRC staff finds that the analysis confirming that the core remains covered is acceptable because by using NRC-approved RETRAN methodology, the licensee determined a minimum margin of 2.7°F so that the turnaround criterion described above is met.

3.4.4.2 Compliance with NRC Imposed Limitations and Conditions

By letter dated November 12, 2024 (Reference 5), the licensee stated that the codes and methods used in the revised non-LOCA analyses are generally the same as those used in the 2012 EPU analyses (Reference 34) with the exception of the transition to Westinghouse Performance Analysis and Design Model (PAD5) and the 3-dimensional RCCA ejection analysis methodologies. In Reference 5, the licensee provided the NRC-approved Turkey Point licensee amendments in which the same methodologies were used as in the proposed non-LOCA analysis. These methodologies are as follows:

- WCAP-7979-P-A, TWINKLE code (Reference 31)
- WCAP-7908-A, FACTRAN code (Reference 32)
- WCAP-14565-P-A, VIPRE code (Reference 25)
- WCAP-11397-P-A, “Revised Thermal Design Procedure” (Reference 23)
- WCAP-14882-P-A and WCAP-14882-S1-P-A, RETRAN-02 code (References 29 and 30)
- WCAP-7907-P-A, LOFTRAN code (Reference 28)
- WCAP-11394-P-A, “Methodology for the Analysis of the Dropped Rod Event” (Reference 35)
- WCAP-15806-P-A, “Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics” (Reference 36)
- WCAP-7588, Revision 1-A, “An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods” (Reference 33)

Based on the above information, the NRC staff finds that all L&Cs listed in the safety evaluations of the above methodologies are satisfied in the proposed non-LOCA analysis.

3.4.4.3 Conclusion

The NRC staff finds the non-LOCA transient analysis acceptable based on the following:

- The licensee used NRC-approved methodologies for the events that were necessary to be reanalyzed due to any proposed changes, as listed in section 3.4.2 of this SE.
- The licensee used acceptable inputs and assumptions for the analyzed events.
- For the events that were not reanalyzed, the licensee justified that they were bounded by other events or were not affected by the proposed changes listed in section 3.4.2 of this SE.
- The results of the events analyzed met the acceptance criteria.

3.4.5 LOCA Containment Analysis

UFSAR, section 14.3.4 describes the current containment integrity analysis. Based on possible change in the fuel decay heat and the stored sensible energy in the reactor internals (for example in fuel assemblies and other components), the fuel transition and fuel cycle transition may impact the following analysis:

- Mass and Energy Release Analyses for Postulated LOCAs (UFSAR, section 14.3.4.1).

- Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures Inside Containment (UFSAR, section 14.3.4.2).
- Containment Response (UFSAR, section 14.3.4.3).
- Available NPSH [net positive suction head] for containment spray pumps, and residual heat removal pumps during LOCA recirculation phase (UFSAR, sections 6.2.3 and 6.3.2).
- Minimum containment pressure analysis for ECCS Performance (UFSAR, section 14.3.2.1.2).

3.4.5.1 LOCA Containment Mass and Energy (M&E) Release Analysis

The LOCA M&E release analysis consists of a short-term and a long-term analysis. The short-term M&E release is used as an input to the containment subcompartment analyses performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse, generally within the first 3 seconds from a double-ended guillotine break of the largest high-energy pipe within that subcompartment. The short-term LOCA M&E releases depend on the break area and the pressure and temperature at the break location. The NRC staff finds it acceptable that short-term LOCA M&E releases and the subsequent subcompartment pressure and temperature response would not be impacted because the break area and the pressure and temperature at the break would not be affected by changes in the fuel loaded in the core or an increase in the fuel cycle length to 24 months.

The licensee analyzed the long-term LOCA M&E release to approximately 10 million seconds and used it as an input to the containment integrity analysis to demonstrate the acceptability of the containment safeguards systems for mitigating the consequences of a postulated large-break LOCA (LBLOCA). The licensee used the NRC-approved WCAP-10325-P-A (Proprietary) (Reference 37) and WCAP-10326-P-A (non-Proprietary) (Reference 37) and associated support review documents (References 38 and 39). The NRC review and approval letters are included in WCAP-10325-P-A (References 37 and 39). Westinghouse has issued Nuclear Safety Advisory Letter (NSAL)-06-6 (Reference 41), NSAL 11-5 (Reference 42), and NSAL 14-2 (Reference 43), which reported several errors in the WCAP-10325-P-A methodology. By letter dated October 3, 2024 (Reference 3), the licensee confirmed that corrected WCAP-10325-P-A methodology with the errors reported in the above NSALs removed was used for the M&E release analysis.

The LOCA M&E release rate cases analyzed by the licensee are for the double-ended pump suction (DEPS) break with **[[**

]]. These LOCA cases are used for the long-term containment response analyses discussed below in section 3.4.5.2 of this SE. The licensee used conservative analysis inputs and included instrumentation uncertainties consistent with the WCAP-10325-P-A methodology and used nominal parameters in certain instances.

The NRC staff finds the long-term M&E release analysis acceptable because the licensee used the NRC-approved methodology while considering the various energy sources described in SRP section 6.2.1.3.

3.4.5.2 LOCA Containment Response Analysis

UFSAR, section 14.3.4 describes the current LOCA and steamline break inside containment response and containment heat removal analyses. These analyses have been performed at conditions related to the 24-month fuel cycle transition using the GOTHIC Version 7.2a computer code. The GOTHIC code is consistent with the NRC-approved evaluation model for R.E. Ginna Nuclear Power Plant (Reference 40). The GOTHIC code, Version 7.2a, was used to take advantage of the diffusion layer model (DLM) heat transfer option. This heat transfer option was approved by the NRC for use in the R.E. Ginna Nuclear Power Plant power uprate containment analyses with the condition that it must be excluded from what was earlier termed as the mist diffusion layer model (MDLM). Since the R.E. Ginna Nuclear Power Plant and Turkey Point both have dry containment designs, the licensee stated that the Turkey Point GOTHIC, Version 7.2a, containment modeling follows the conditions of acceptance placed on the R.E. Ginna Nuclear Power Plant and is consistent with the restrictions identified in Reference 40. The licensee further stated that none of the user-controlled enhancements added to GOTHIC, Version 7.2a were implemented in the Turkey Point containment model.

[[
]], the licensee stated that the current analysis of the post-LOCA long-term decay heat removal is not impacted for the proposed change to the 24-month cycle with new fuel.

The acceptance criteria for the containment response are that the containment safeguards systems shall be capable of limiting the peak containment pressure to less than the containment design pressure of 55.0 psig and the containment wall temperature to less than the containment structural design temperature of 283.0°F. In addition, the safeguards systems must also be capable of reducing the peak pressure at 24 hours to half of the calculated peak value.

The licensee's analysis addressed a spectrum of cases based on break location along with postulated single failure. The limiting break is the double-ended pump suction break with **[[** **]]**. For this case, the results are as follows:

- Peak pressure increased from its current value of 53.85 psig to 53.94 psig.
- Pressure at 24 hours increased from its current value of 14.4 psig to 14.84 psig.
- Peak wall temperature increased from 273.5°F to 273.7°F.

The NRC staff finds that the results of the containment response analysis for the 24-month fuel cycle with new fuel is acceptable because the results are bounded by the acceptance criteria of containment design pressure of 55 psig and its structural design of 283.0°F. The pressure of 14.84 psig at 24 hours from initiation of LOCA is also bounded by 26.97 psig, which is half of the peak pressure and, therefore, acceptable.

3.4.5.3 Net Positive Suction Head (NPSH) Analysis

A change in the sensible and decay heat due to the proposed fuel cycle change and fuel transition to new fuel affects the M&E release in the containment during a large break LOCA and, therefore, affects the containment sump temperature response. The important parameter is the available NPSH (NPSHA) for the pumps that draw water from the containment sump during the LOCA recirculation phase, i.e., the high-head safety injection (HHSI) pumps and the containment spray system (CSS) pumps because of the high sump water temperature at the initiation of this phase. The NPSHA of these pumps depends on the vapor pressure at the sump temperature, static head at the pump inlet, head loss in the pump inlet piping, and the containment accident pressure (CAP) above the sump. The licensee used the following assumption in the calculation of NPSHA:

- Pressure above sump is assumed to be the vapor pressure at sump temperature.
- **[[ ]]**.
- Pump inlet piping head loss is calculated at the maximum flow condition.

By letter dated October 3, 2024 (Reference 3), the licensee provided the results of NPSHA, required NPSH (NPSHR), and NPSH margin (NPSHA – NPSHR) given in table 11 below.

Table 11: NPSH Results

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3.4.5.4 Minimum Containment Pressure Analysis for ECCS Performance

The licensee stated that the containment pressure calculation for the ECCS analysis was performed consistent with the NRC-approved methodology (refer to Section 3.4.3.5 of this SE for the licensee's compliance statement to L&C #3). Appropriate design parameters and conditions were modeled, as were the engineered safety features that can reduce the containment pressure.

Reference 1, Table 4.8-2, item 3, "Containment Parameters," provides the containment pressure used for Region I and Region II LOCA analysis. For Region I, the licensee used constant containment pressure equal to its initial pressure given in UFSAR Table 14.3.2.1-2. For Region II, the licensee calculated the containment pressure using transient specific mass and energy releases for each event and the information on containment data in Reference 1,Tables 4.8-3, 4.8-4, and 4.8-5.

The NRC staff determined that the minimum containment pressure used for ECCS performance analysis is acceptable because of the following: for Region I analysis, the licensee used constant pressure equal to the initial containment pressure, which is conservative and for Region II analysis, the licensee used the NRC-approved methodology using appropriate containment design parameters that reduce the containment pressure.

3.4.5.5 Conclusions

Technical Conclusions

- The NRC staff finds that the results of the containment response analysis for the 24-month fuel cycle with new fuel is acceptable because the LOCA peak containment pressure and temperature are bounded by the containment design pressure and structural design temperature, respectively.

- The NRC staff finds that the licensee's results of the NPSH analysis for the RHR pumps that draw water from the sump during the LOCA recirculation phase are acceptable because the minimum NPSH margin during the transient is positive.

- The NRC staff finds that the minimum containment pressure analysis for ECCS performance is acceptable because for Region I analysis, it conservatively uses the initial containment pressure and for Region II analysis, the pressure is conservatively calculated using NRC-approved codes and containment input parameters that reduce the pressure.

Regulatory Conclusions

Based on the above technical conclusions, the NRC staff finds that the following 1967 Proposed GDC are satisfied:

- GDC 10 is satisfied because containment integrity is maintained under the most limiting large break LOCA conditions in the presence of limiting single active failure.

- GDC 41 is satisfied because the ECCS and containment heat removal systems sufficiently perform to fulfil the required safety function assuming a limiting single active component.

- GDC 49 is satisfied because the containment heat removal systems can accommodate without exceeding the design leakage rate resulting from the largest credible energy release following a large break LOCA since the peak pressure and temperature and the pressure at 24 hours from a large break LOCA are bounded by the accepted design values.

Based on the technical and regulatory conclusions described above, the NRC staff finds that the licensee's containment analysis for fuel cycle transition with proposed new fuel is acceptable.

3.4.6 Steamline Break Containment Analysis

UFSAR, section 14.3.4 describes the current containment pressure response to steamline break (SLB). Consistent with the current licensing basis, the licensee used the RETRAN code for the SLB M&E release analysis and the GOTHIC computer code for containment response. The licensee analyzed M&E releases for various initial power levels, break definitions, and single-failure assumptions. The analysis credited full containment safeguards of two fan coolers and two containment spray pumps for the single failure assumption of auxiliary feedwater (AFW) pump runout protection failure case, feedwater isolation valve failure case, and the main steamline check valve (MSCV) failure case. For the analysis case of failure of one train of safeguards due to the breakdown of an emergency diesel generator, the licensee credited operation of only one fan cooler and one containment spray system train. The most limiting results of the licensee's analysis are as follows:

- Peak pressure is 53.04 psig for a 1.4 ft^2 DER at HZP assuming an MSCV single failure.

- Peak wall temperature is 278.9°F for a 1.4 ft^2 DER at HZP assuming an AFW runout protection single failure.

The NRC staff finds that the results of the SLB containment response analysis for the 24-month fuel cycle transition with the new fuel are acceptable because they are bounded by the acceptance criteria of containment design pressure of 55 psig and its structural design temperature of 283.0°F.

3.4.7 Component Cooling Water (CCW) Analysis

As described in UFSAR, section 9.3.2, the design basis of the CCW system is to transfer an adequate amount of heat from the ESF systems to the ultimate heat sink (UHS) during post-accident operation while considering the most limiting single active failure. For the proposed fuel cycle transition with new fuel, the licensee analyzed the post-accident operation because the peak CCW system operating temperatures occur during this scenario due to higher heat loads and heat rejection into the system. The calculated peak temperature is used as an input in analyses that demonstrate the acceptability of the CCW system post-accident pipe stresses. The licensee used the NRC-accepted GOTHIC computer code for the post-accident thermal analysis. The licensee stated that the CCW GOTHIC model parameters, which include containment volume, initial conditions, containment heat sinks, mass and energy releases, and the containment spray system parameters, are consistent with the containment response model described in section 3.4.2.2 of this SE. The licensee incorporated the following changes in the proposed LOCA and SLB analysis from the current analysis:

- Updated containment M&E release.
- Increased the intake canal water (ICW) temperature from 100°F to 104°F
- Updated the CCW heat exchanger fouling factor consistent with the ICW temperature increase because the fouling increases with increasing temperatures.
- Updated containment heat sink surface areas for LOCA containment response only.

The licensee made the following key assumptions for calculating the maximum CCW temperature:

- Maximized heat transfer from containment for conservatively maximizing the CCW temperature even though for conservative containment integrity analysis heat transfer to the CCW system is minimized. The CCW temperatures are maximized using the following assumptions:

  - Fan cooler heat removal capacity is maximized resulting in a larger heat load on the CCW system prior to LOCA recirculation phase.

  - Maximized CCW flowrate as this results in a larger heat load on the CCW system which results in the most conservative CCW system temperature response.

- For the LOCA scenario, the licensee considered three single failure cases, which are: diesel generator failure, failure of one containment spray pump, and failure of one ICW pump.

- For the SLB, the CCW limiting temperature response is based on the data from the SLB containment integrity analysis which determined that the 1.4 ft^2 DER initiated at HZP assuming an AFW runout protection failure M&E release.

The licensee's results for the current CCW performance parameters (heat loads and temperatures) from the GOTHIC analysis for the SLB are bounded by the current LOCA analysis results. Also, the results from the revised SLB GOTHIC analysis indicates CCW system performance parameters are bounded by the current SLB analysis.

By letter dated October 3, 2024, the licensee provided the current and revised CCW performance peak temperatures based on LOCA GOTHIC analysis as shown in table 12 below.

<u>Table 12: CCW Performance Results from the LOCA GOTHIC Analysis</u>

Parameter	**Current Peak Values (°F) LOCA**	**Revised LOCA Peak Values (°F)**
CCW Supply Temperature	158.6	159.5
CCW Return Temperature	185.4	186.0
ECC Outlet Temperature (RHR heat exchanger (Hx) inlet temperature)	205.4	205.9
RHR Hx Outlet Temperature	193.0	194.6

From the above table, the CCW performance results from the revised LOCA analyses indicate small increase in the peak CCW temperatures from their current LOCA analysis results. Also, the revised LOCA analysis results bound both the current and revised SLB analysis results.

Based on the above results the licensee stated the following:

- With the small increase from 185.4°F to 186°F, the CCW piping temperature still remains below the analyzed limit used in the current CCW pipe stress analysis.

- Because of the small increase in the CCW return temperature (185.4°F to 186°F), the CCW vapor pressure will slightly increase and, therefore, affects the CCW pump NPSH margin. The current NPSHA is 122.3 ft and the CCW pump NPSHR is 46 ft. Since there is a large NPSH margin, there will still remain a significant NPSH margin.

- The heat load per CCW heat exchanger increases by 1.4 MBtu/hr, which is accommodated by the existing margin of 2.6 MBtu/hr.

The NRC staff finds the licensee's evaluation of CCW post-accident operation under the proposed fuel transition and fuel cycle change conditions acceptable because of the following:

- By using NRC-accepted GOTHIC methodology, the licensee demonstrated that the small increase in CCW peak temperatures due to increased heat load remains bounded by the temperatures used for CCW piping stress analysis.

- Each CCW heat exchanger has sufficient design margin to accommodate the increased post-accident heat load.

3.4.8 Steam Generator Tube Rupture

This SG tube rupture (SGTR) event is assumed to be caused by the instantaneous complete rupture of an SG tube releasing primary coolant to the lower pressure secondary system. The major hazard associated with this event is the potential radiological consequences resulting from the release of radioactive reactor coolant to the secondary side of the ruptured SG and subsequent release of radioactivity to the atmosphere. Another major hazard to the SGTR event is overfilling of the ruptured SG, which would invalidate the assumption used in the radiological analysis because this analysis assumes that SG overfill does not occur prior to break flow termination.

Consistent with the current licensing basis, for the proposed change to 24-month fuel cycle with new fuel, the licensee performed thermal-hydraulic analysis for the SGTR event and calculated steam release as an input to the radiological dose analysis. UFSAR, section 14.2.4 describes the following SGTR licensing basis thermal-hydraulic analysis:

- Steam release analysis by hand calculation.

- Thermal-hydraulic margin to overfill (MTO) analysis using LOFTTR2 code

- Confirmatory thermal-hydraulic steam release analysis using LOFTTR2 code for dose calculation.

LOFTTR2 is a Westinghouse thermal-hydraulic code developed to perform STGR steam release and MTO analysis. An earlier version of this code, LOFTTR1, was developed as part of NRC-approved TR WCAP-10698-P-A/WCAP-10750-A and their Supplement 1 (References 44 and 45) analysis methodology. The licensee revised this code to the LOFTTR2 version for a more realistic MTO analysis. The licensee stated that the two versions are identical except that the LOFTTR2 version includes an additional capability to represent the transition from two regions (steam and water) in the SG secondary side to a single water region if overfill occurs, and transition back to two regions depending on the SG secondary side conditions. The NRC SE of TR WCAP-10698-P-A/WCAP-10750-A includes the evaluation of change to LOFTTR2. The NRC staff finds the use of the LOFTTR2 code acceptable because it is identical to the NRC-approved LOFTTR1 with the additional capability of a realistic MTO analysis, which has been approved by the NRC SE of TR WCAP-10698-P-A/WCAP-10750-A.

As stated in UFSAR, section 14.2.4, the current analysis assumes that the termination of the ruptured SG activity release occurs when the ruptured SG is isolated at 30 minutes by operator action. While this isolation terminates releases from the ruptured SG, primary-to-secondary leakage continues to provide activity for release from the unaffected SGs.

Consistent with the current licensing basis, for the proposed 24-month fuel cycle with new fuel, the licensee used the LOFTTR2 code to confirm that the SGTR break flow terminates at 30 minutes. The licensee also performed MTO analysis to demonstrate that the shell of the SG with the ruptured tube does not overfill during the accident.

By letter dated October 3, 2024 (Reference 3), the licensee provided the following information regarding the current and the proposed SGTR hand-calculation analysis for the 24-month cycle transition with the new fuel:

- A complete tube break adjacent to the tube sheet was considered in the current and the proposed analysis.

- A comparison of key input values used in the proposed analysis and the current analysis.

The NRC staff notes that all inputs in both analyses are the same except for the following:

(a) SG tube plugging conservatively increased from 10 percent in the current analysis to 15 percent in the proposed analysis,

(b) the maximum AFW temperature conservatively increased from the current analysis value of 100°F to 107°F in the proposed analysis to address the temperature rise across the AFW pumps,

(c) the AFW actuation delay following low pressurizer pressure SI decreased from 120 seconds to 95 seconds. As stated in letter dated October 3, 2024, the 95 seconds is plant-specific for Turkey Point, whereas the 120 seconds was used in the current analysis (2012 EPU) as a conservative input, and

(d) the decay heat model, 1971 ANS +20 percent, is the same, but conservatively increased in the proposed analysis to account for gadolinia.

By letter dated October 3, 2024, the licensee provided steam release mass flow rate results obtained by hand-calculation analysis for the 24-month cycle with the new fuel and their comparison with the current analysis results given in UFSAR, table 14.2.4-2. The comparison shows that they are either the same or bounded by the current analysis results.

The results of the licensee's MTO analysis for the 24-month cycle with the new fuel using the LOFTTR2 code demonstrate that the break flow can be terminated before overfilling of the ruptured SG occurs.

The results of licensee's confirmatory steam release analysis using the LOFTTR2 code for the 24-month cycle with the new fuel demonstrated that, despite the continuation of break flow beyond the 30-minute termination time assumed in the steam release analysis for input to the licensing basis radiological dose consequences analysis, the mass transfer data calculated in the 30-minute steam release for dose analysis is bounding. Therefore, the current licensing basis radiological consequences analysis will remain bounding.

The NRC staff finds that the evaluation of the SGTR event for the 24-month cycle with the new fuel is acceptable because of the following:

(a) consistent with the current licensing basis, the licensee used hand-calculation for determining the steam release rates, which are inputs to the radiological consequence analysis,

(b) the MTO analysis result based on NRC-approved LOFTTR2 computer code is acceptable as the SG does not overfill during the SGTR event, and

(c) the results of the confirmatory steam release analysis performed by the NRC-approved LOFTTR2 computer code show that the mass transfer data calculated in the 30-minute steam release for dose analysis is bounding, which implies that the current licensing basis radiological consequences analysis for the SGTR event will remain bounding.

3.4.9 Neutron Fluences

The licensee performed a discrete ordinates (Sn) transport analysis to evaluate the change in the neutron radiation environment for the materials comprising the beltline and extended beltline regions of the reactor pressure vessel (RPV) due to the proposed transition to 24-month cycle designs. The RPV neutron exposure projections were determined using the past 18-month cycle designs for Turkey Point, Unit No. 3 and two representative 24-month cycle designs. The licensee noted that that while the past cycle designs considered in the analysis are only applicable to Turkey Point, Unit No. 3, the overall results are applicable to both Unit Nos. 3 and 4 since both units have the same RPV design and utilize the same fuel designs and fuel management strategies. Further, both units are operated in a similar manner and have an integrated surveillance program.

3.4.9.1 Neutron Fluence Analysis

The licensee stated that the radiation transport methodology used for the neutron exposure analysis follows the guidance of RG 1.190 (Reference 52). The methodology used is consistent with the NRC-approved methodology described in WCAP-18124-NP-A (Reference 46) for the

RPV beltline region (i.e., in general, RPV materials opposite the active fuel) and WCAP-18124-NP-A, Revision 0, Supplement 1-NP-A (Reference 47) for the RPV extended beltline region.

The licensee utilized the previous in-vessel surveillance capsules that have been withdrawn from the reactor core and that were analyzed as part of the reactor vessel materials surveillance program. From the capsule data, the licensee noted that only the data from capsules T and X were used since the measurement data for the sensors in capsules S and V were inconsistent with the normalized data for other 3-loop thermal shield plants. The results of the plant-specific transport calculations and capsules T and X dosimetry evaluations were used to demonstrate the ±20 percent (1σ) uncertainty between measured and calculated data, per RG 1.190 guidance.

3.4.9.2 Results

The neutron fluence projections for Turkey Point, Unit Nos. 3 and 4 for the subsequent license renewal application (SLRA) are documented in the NRC safety evaluation report (SER) (Reference 53). The licensee stated in the LAR that the exposure analysis performed showed that the fast neutron (E>1.0 MeV) fluence projections for the RPV and the RPV welds were greater than or essentially equal to the corresponding 24-month cycle values. Based on this, the licensee concluded that neutron exposure projections do not need to be updated to account for the transition to 24-month cycles. The NRC staff found the neutron fluence analysis to be acceptable based on the use of NRC-approved methodology and guidance to perform the evaluations.

3.4.9.3 Compliance with NRC Imposed Limitations and Conditions

The NRC staff evaluation of compliance with the L&Cs listed in the SEs of the TRs used in the neutron fluence analysis is given below.

WCAP-18124-NP-A

L&C #1

This L&C states that the applicability of WCAP-18124-NP, Revision 0, is limited to the RPV region near the active height of the core based on the uncertainty analysis performed and the measurement data provided.

Compliance with L&C #1

The conditions necessary to meet the L&C #1 are provided in WCAP-18124-NP-A Revision 0, Supplement 1-NP-A, Revision 0, which allows for application of RAPTOR-M3G methodology to the RPV extended beltline region on a generic basis. The licensee stated that it used Supplement 1-NP-A for the extended beltline region.

L&C # 2

Least squares adjustment is acceptable if the adjustments to the M/C ratios and the calculated spectra values are within assigned uncertainties of the calculated spectra, the dosimetry measured reaction rates, and the dosimetry reaction cross Sections. Should this not be the case, the user should re-examine both measured and calculated values for possible errors. If errors cannot be found, the particular values causing inconsistency should be disqualified.

Compliance with L&C #2

The licensee did not indicate the use of the least squares analyses to adjust any calculated RPV or surveillance capsule neutron exposures. Further, the results of the plant-specific transport calculations and capsule dosimetry evaluations had the uncertainty between measured and calculated data within the values provided by RG 1.190.

Based on its review of the LAR, the NRC staff finds that all the applicable limitations and conditions from the TRs listed in the neutron fluence analysis have been met.

3.4.9.4 Conclusion

The NRC staff finds the neutron fluence analysis to be acceptable since the fluence values for projected 72 effective full-power years (EFPYs) performed for the Turkey Point SLRA are applicable to the 24-month cycle transition given that there is minimal change in the fuel design to impact fluence projections and the licensee validation of measured and calculated data using the surveillance capsule evaluations.

3.4.10 Fuel Rod Design and Performance

The licensee conducted analyses of fuel rod design to evaluate how the potential effects in fuel design and the transition to 24-month fuel cycle operating conditions could impact meeting the fuel rod design criteria for Turkey Point.

The fuel rod performance for all Turkey Point fuel has been demonstrated to meet the fuel rod design bases (Reference 49) in the SRP. Turkey Point UFSAR, Chapter 3 summarizes the current fuel design and application. The reactor core is currently a three-region cycled core. These same bases are applicable to all fuel rod designs, including the Westinghouse 15x15 Upgrade PRIME fuel design with AXIOM fuel cladding, ADOPT fuel pellets, and gadolinia (Gd_2O_3) doped uranium dioxide (UO_2) fuel pellets.

The PAD5 code, approved by the NRC along with its specified models WCAP-18546-P-A, WCAP-18482-P-A, and WCAP-17642-P-A (References 20, 21, and 22, respectively) for in-reactor behavior, serves to compute the fuel rod performance throughout its irradiation history. PAD5 functions as the primary tool for assessing fuel rod performance. COROSN employs identical thermal, corrosion, and hydriding models as PAD5 but is optimized for effectively evaluating oxidation and hydriding design requirements.

The evaluation of transitioning to the 15x15 Upgrade PRIME fuel design, incorporating AXIOM fuel cladding along with ADOPT and GAD fuel pellets, included assessments of mixed core configurations combining the existing 15x15 Upgrade fuel design with the new 15x15 Upgrade PRIME design, as well as evaluations of full cores utilizing the 15x15 Upgrade PRIME fuel assembly design.

The criteria pertinent to the fuel rod design are rod internal pressure (RIP), cladding corrosion, cladding stress and strain, cladding fatigue, plenum cladding support, fuel rod axial growth, cladding flattening, cladding free standing, and fuel centerline melt.

As expected, the extension of cycle length, changes in void volume, and introduction of new fuel types affect the available operating margin in the evaluation of fuel rod design. Criteria such as RIP, transient cladding strain, cladding fatigue, clad stress, and fuel centerline melt have experienced margin reductions. Despite this, all limits were still adhered to during the analysis of the 24-month fuel cycle transition. Each of these fuel rod design criteria will be verified on a cycle-specific basis as part of the reload safety evaluation following the Westinghouse reload methodology (Reference 19). The applicability of the Westinghouse 15x15 Upgrade PRIME fuel design at Turkey Point, Unit Nos. 3 and 4 under the conditions of the 24-month fuel cycle transition has been evaluated for each of these key fuel rod design criteria. Based on these assessments, the NRC staff concludes that all design criteria can be satisfied for the 24-month fuel cycle transition.

3.4.11 Mechanical Compatibility and Performance

The NRC staff evaluated the mechanical compatibility and performance for transitioning to the 15x15 Upgrade PRIME fuel design, incorporating AXIOM fuel cladding, and ADOPT and GAD fuel pellets to support the 24-month cycle project. This evaluation encompassed mixed core configurations involving both the existing 15x15 Upgrade fuel design and the new 15x15 Upgrade PRIME fuel designs. It specifically addresses the impact on fuel assembly (FA) performance and evaluates two aspects of fuel rod performance.

The criteria pertinent to the mechanical compatibility and performance of the FA that must be satisfied are the FA top nozzle holddown force, FA shipping and handling loads, seismic/LOCA analysis, and FA interfaces, clearances, and compatibility. Criteria pertinent to the compatibility and performance of the fuel rod that must be satisfied are grid to rod fretting wear (GTRF) and fuel rod shoulder gap. How these criteria are satisfied is discussed below.

3.4.11.1 Fuel Assembly Performance

The transition to the 15x15 Upgrade PRIME fuel design results in a slightly lower pressure drop and consequently a slightly higher estimated flow rate in the RCS. Additionally, there is a minor increase in the fuel assembly weight (approximately 1 percent) due to the adoption of ADOPT and GAD fuel pellets. Both of these changes could potentially affect the holddown force at the top nozzle of the FA. However, an evaluation has confirmed that these changes are not significant and that all criteria for top nozzle holddown force are satisfied.

The mechanical design criteria for shipping and handling loads during Condition I and II events, as well as various structural design criteria for FAs, specific to the 15x15 Upgrade PRIME fuel product, were comprehensively assessed. It was determined that all acceptance criteria related to FA structural design requirements are met.

The structural integrity of the 15x15 Upgrade PRIME bottom nozzle is equivalent to that of the current/resident 15x15 Upgrade Debris Filter Bottom Nozzle (DFBN). Analysis of bounding grid impact sensitivities for Seismic/LOCA events (Condition II, III, and IV) using PRIME fuel with Low Tin ZIRLO grid properties confirmed that the grid impact assessment, guide tube stress evaluation, and fuel rod stress results from the primary analysis remain bounding and applicable.

All relevant interfaces and clearances remain valid and bounding, as the FA envelopes and interfacing dimensions have not changed in the transition to the 15x15 Upgrade PRIME fuel design compared to the current 15x15 Upgrade fuel design. Therefore, all applicable interfaces and clearances remain acceptable and are compatible with the existing 15x15 Upgrade fuel design.

3.4.11.2 Fuel Rod Performance

For grid to rod fretting wear (GTRF), the criterion that needs to be met is that wall thickness reduction be no greater than 10 percent when evaluating cladding imperfections, including fretting wear marks.

The transition to Low Tin ZIRLO grids will decrease the oxide layer thickness (reducing corrosion rates) and minimize grid growth, both of which contribute to lower GTRF levels. Reduced grid growth results in a wider gap between the grids and fuel rods, thereby lowering the potential for fretting. Overall, the adoption of Low Tin ZIRLO grids is expected to provide increased margin against GTRF compared to the current 15x15 Upgrade design. The thinner oxide layers associated with AXIOM fuel rod cladding also provide ample protection against GTRF. Wear is anticipated to be significantly below the 10 percent criterion for the 15x15 Upgrade PRIME fuel design with AXIOM cladding at Turkey Point. Therefore, the risk of GTRF failure is considered very low and acceptable.

The criterion for Fuel Rod Shoulder Gap requires that the space between the top of the fuel rod and the bottom of the top nozzle adapter plate be sufficient to prevent their contact. There have been no changes to either the grid or fuel rod designs that would hinder the grid’s ability to accommodate differential expansion between the fuel rods and their positions within the fuel assembly skeleton. Furthermore, the tube-in-tube guide thimble design enhances overall fuel stiffness. Thus, the criteria regarding fuel assembly distortion and fuel rod buckling are met, as it has been demonstrated that the 15x15 Upgrade PRIME fuel assembly design, featuring AXIOM fuel cladding and ADOPT and GAD fuel pellets, provides more than adequate shoulder gap. In conclusion, the criteria related to fuel rod shoulder gap, fuel assembly distortion, and fuel rod buckling are met.

3.4.11.3 Cumulative Effects of Fuel Changes

The Turkey Point 24-month cycle project incorporates several fuel design modifications, including the 15x15 Upgrade PRIME fuel design, AXIOM fuel cladding, and ADOPT and GAD fuel pellets. The assessment of these changes considered the integrated and cumulative effects on fuel assembly weight, grid crush values, corrosion rates (oxide layer), component material properties, fuel rod growth rates, fuel assembly/grid dimensional performance, and GTRF performance. All these factors have been collectively evaluated to ensure compliance with all fuel mechanical design criteria. Based on the evaluations presented, the NRC staff determined that the cumulative effects of the Turkey Point 24-month cycle project fuel design changes on the relevant fuel mechanical design criteria are acceptable and that all criteria are satisfied because the licensee uses NRC-approved Upgrade PRIME fuel design, AXIOM fuel cladding WCAP-18546-P-A, and ADOPT and GAD fuel pellets WCAP-18482-P-A, and the evaluation of the cumulative effects of those changes have been assessed to comply with all fuel mechanical design criteria.

3.4.11.4 Technical Conclusion

Regarding the mechanical design changes associated with the 15x15 Upgrade PRIME fuel assembly design, including AXIOM fuel cladding and ADOPT and GAD fuel pellets, the NRC staff concludes that this integrated fuel design is structurally and mechanically acceptable, meeting all applicable design and safety criteria. The 15x15 Upgrade PRIME fuel design, with AXIOM fuel cladding and ADOPT and GAD fuel pellets, is compatible with the existing 15x15 Upgrade fuel design.

Based on its review of LAR, the NRC staff concludes that the Turkey Point 24-month fuel cycle transition project fuel design changes on the applicable fuel mechanical design criteria are acceptable and that all criteria remain satisfied.

3.4.12 Control Rod Ejection

The licensee applied the control rod ejection methodology specified in WCAP-15806-P-A, “Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics” (Reference 36). This methodology was approved by the NRC in 2003 and reviewed in accordance with RG 1.77, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors” (Reference 54). Since the approval of WCAP-15806-P-A, the NRC has rescinded RG 1.77 (Reference 55) and issued RG 1.236, “Pressurized -Water Reactor Control Rod Ejection and Boiling Water Control Rod Drop Accidents” (Reference 56). The primary differences between RG 1.77 and RG 1.236 are in the fuel rod failure thresholds and the limits on damaged core coolability. The new failure thresholds and limits on damaged core coolability do not invalidate the acceptability of the WCAP-15806-P-A method to calculate the relevant phenomena discussed in the TR such as enthalpy rise, peak fuel enthalpy rise, departure from nucleate boiling, fuel temperature, and pressure surge. WCAP-15806-P-A was also applied in WCAP-17524-P-A, Revision 1, “AP1000 Core Reference Report” (Reference 57), and reviewed in accordance with interim guidance for reactivity initiated accidents in Appendix B of Revision 3 to Chapter 4.2, “Fuel System Design,” of NUREG-0800 (Reference 49), which predated RG 1.236. Additionally, L&C 1 of the ADOPT pellet TR, WCAP-18482-P-A, relates to the modeling of control rod ejection accidents. Specifically, it states:

> Licensees must demonstrate that the CRE analytical models, methods, and acceptance criteria are applicable to fuel designs containing ADOPT pellets and capture all relevant fuel burnup and cladding corrosion related phenomena

As such, Enclosure 4 to the LAR provides plant specific justification for the application of WCAP-15806-P-A to address RG 1.236 and details how the relevant fuel burnup and cladding corrosion phenomena are captured. The licensee stated that the NRC-approved Westinghouse fuel performance PAD5 code, as described in WCAP-17642-P-A, Revision 1, “Westinghouse Performance Analysis and Design Model (PAD5)” (Reference 22), was used to generate inputs to WCAP-15806-P-A and to translate the rod ejection failure thresholds to functions of burnup. The high temperature failure threshold (peak radial average enthalpy as a function of cladding pressure differential) in figure 1 of RG 1.236 was converted to a burnup dependent limit using steady-state rod internal pressure conservatively calculated from PAD5. The steady state rod internal pressure was adjusted to account for transient fission gas release using the transient fission gas release correlations in Appendix B of RG 1.236. PAD5 has been approved to calculate rod internal pressure in WCAP-17642-P-A and transient fission gas release is appropriately accounted for. WCAP-15806-P-A was used to evaluate if any rods fail due to experiencing DNB. The licensee calculated that no rods were expected to fail due to high temperature phenomena. The NRC staff finds this application of the high temperature cladding failure threshold and the evaluation of high temperature failure phenomena acceptable because it uses NRC-approved models to account for rod internal pressure combined with WCAP-15806-P-A, which has been approved to calculate fuel enthalpy and the DNBR during a rod ejection accident.

For the pellet-cladding mechanical interaction (PCMI) phenomena, section 3.2 and figures 2 through 5 of RG 1.236 present failure thresholds for stress-relieved annealed (SRA) cladding and recrystallized annealed (RXA) cladding that are in terms of peal radial average enthalpy rise as a function of cladding excess hydrogen content. Since AXIOM cladding has a partially RXA (pRXA) heat treatment, the PCMI failure thresholds in RG 1.236 do not apply. As a result, the licensee applied the NRC-approved PCMI failure threshold for AXIOM detailed in section 3.10.1 of WCAP-18546-P-A, specifically by using conservative values for hydrogen as a function of burnup that is implemented in WCAP-15806-P-A. Section 5.2 of WCAP-18546-P-A details the NRC-approved hydrogen pickup model that is used with PAD5. The licensee calculated that no rods were expected to fail due to PCMI. The NRC staff finds this application of the PCMI failure threshold and the evaluation of PCMI to be acceptable because it uses the AXIOM-specific failure threshold and an NRC-approved hydrogen pickup model combined with the WCAP-15806-P-A method to calculate fuel enthalpy rise.

Concerning the other RG 1.236 criteria, the NRC-approved PAD5 fuel melt question was used to assess if any fuel melt was anticipated to occur. No fuel melt was calculated to occur in the Turkey Point rod ejection analysis, so the criteria in section 3.3 and section 6 of RG 1.236 are satisfied. The pressure surge was calculated to be less than Turkey Point's reactor coolant system pressure limit with WCAP-15806-P-A and thus meets section 5 of RG 1.236. Since no rods were calculated to fail by the various phenomena described above, section 4 of RG 1.236 is satisfied.

The NRC confirmed that the Turkey Point plant-specific analysis considered the full range of operation from beginning-of-cycle to end-of-cycle and that the analysis assumptions were conservative.

Overall, the NRC staff finds that the application of WCAP-15806-P-A combined with PAD5 to Turkey Point is acceptable because the method and analysis meet the applicable regulatory positions of RG 1.236. Particularly, the method and analysis acceptably consider the various failure thresholds and limits discussed in RG 1.236 and appropriately capture relevant fuel burnup and cladding corrosion related phenomena (and thus also satisfy L&C 1 of WCAP-18482-P-A).

3.4.13 Non-LOCA Gap Release Fractions

Sections 4.16 and 4.18 of the LAR state that the design basis accident radiological consequences analyses were evaluated for the following non-LOCA DBAs: locked rotor, fuel handling accident, accidental release – waste gas, steam generator tube rupture, main steam line break, and rod ejection. The licensee concluded that the design basis analyses for these events reported in the UFSAR are bounding of the dose analyses performed to support the 24-month cycle transition and fuel transition.

The transition from 18- to 24-month cycles would not be expected to impact the fuel-cladding gap release fractions if the burnup and linear heat generation rate thermal mechanical operating limit remain constant, though the implementation of dopants and larger grain boundaries in the fuel pellet, which are characteristics of the ADOPT fuel pellet, has the potential to impact the fission product fuel-cladding gap release fractions. Section 6.2.4 of WCAP-18482-P/NP-A, “Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel,” and section 3.6.3.4 of the staff’s SE of WCAP-18482-P/NP-A discuss the effects of the ADOPT fuel pellet on the gap release fractions and radiological consequence analysis. Section 6.2.4 of WCAP-18482-P/NP-A concludes that the methods currently used in the analyses of radiological consequences of design basis accidents are valid for the ADOPT fuel pellet design.

Based on the above, the use of the current non-LOCA gap release fractions within the existing radiological consequence analyses of record is consistent with WCAP-18482-P/NP-A and acceptable.

3.5 Evaluation of SR 3.6.3.5 Proposed Change

Containment purge supply and exhaust valves (or purge valves) are containment isolation valves and, therefore, leakage testing of these valves must satisfy the leakage test requirements of 10 CFR Part 50, Appendix J. Additionally, the leakage test frequency for these valves is established by CFR Part 50, Appendix J. The licensee’s Containment Leakage Rate Testing Program (CLRTP) administers the 10 CFR Part 50, Appendix J program for Turkey Point.

The licensee proposed a change that would revise the surveillance frequency specified in SR 3.6.3.5 from the Surveillance Frequency Control Program (SFCP) to the CLRTP of TS 5.5.13. The NRC staff reviewed the proposed change and finds that the proposed modification for SR 3.6.3.5 is both necessary and accurate as the containment isolation purge valves are tested in accordance with the CLRTP and not the SFCP. The proposed change will ensure that the requirement that the purge valves are tested in accordance with 10 CFR Part 50, Appendix J continues to be met and, therefore, the change is acceptable.

3.6 Evaluation of TS 5.6.3 Proposed Changes

Turkey Point TS 5.6.3.b lists the NRC-approved analytical methods used to determine the Turkey Point core operating limits. In LAR section 2.0, “Background,” the licensee described the use of AXIOM cladding. With the use of AXIOM cladding, the licensee proposed changes to TS 5.6.3.b to include WCAP-18546-P-A, “Westinghouse AXIOM Cladding for Use in Pressurized Water Reactor Fuel,” in the list of approved COLR references because WCAP-18546-P-A is used to determine core operating limits.

Based on the technical evaluation in Section 3.0 of this SE, the NRC staff concluded that the use of AXIOM cladding is acceptable. The NRC staff determined that the proposed change to TS 5.6.3 is acceptable because WCAP-18546-P-A is an analytical method used to determine the core operating limits that has been reviewed and approved by the NRC and the proposed change to TS 5.6.3 is consistent with guidance contained in the Westinghouse standard technical specifications (NUREG-1431).

The proposed change would also remove six Turkey Point TS 5.6.3.b COLR references and two associated notes for determining $F_Q(Z)$, $F_\Delta H$, and the K(Z) curve. In LAR section 2.0, the licensee explained that these six COLR references are no longer utilized due to full implementation of the WCAP-16996-P-A (FSLOCA) methodology. Based on the full implementation of WCAP-16996-P-A, the NRC staff concludes that the six COLR analytical methods are no longer used and can be deleted. Likewise, the information contained in the two associated notes is no longer applicable (outdated) and can be deleted as well. Renumbering the remaining references consistent with these deletions is appropriate and does not impact any applicable requirements.

Based on the above, the NRC staff finds that with the proposed changes, 10 CFR 50.36(c)(5) will continue to be met because TS 5.6.3 continues to provide administrative controls necessary to assure operation of the facility in a safe manner.

3.7 Evaluation of TS 4.2.1 Proposed Changes

Turkey Point TS 4.2.1 provides descriptive information on fuel assembly design features such as fuel rod cladding and fuel material. In LAR section 2.0, the licensee described the proposed use of AXIOM cladding and ADOPT fuel. With the proposed use of AXIOM cladding and ADOPT fuel, the licensee also proposed changes to TS 4.2.1 to reflect these new design features associated with fuel assembly design.

Based on the technical evaluation in Section 3.0 of this SE, the NRC staff concluded that the use of AXIOM cladding and ADOPT fuel is technically acceptable. The NRC staff determined that the proposed change to TS 4.2.1 is acceptable because it contains information that reflects the proposed change to the fuel assembly design features and is consistent with guidance contained in the Westinghouse standard technical specifications (NUREG-1431). Therefore, the NRC staff finds that with the proposed changes, 10 CFR 50.36(c)(4) will continue to be met because TS 4.2.1 continues to include design features of the facility, which, if altered or modified, would have a significant effect on safety and are not covered elsewhere in 10 CFR 50.36.

4.0 STATE CONSULTATION

In accordance with the Commission’s regulations, the Florida State official was notified of the proposed issuance of the amendments on January 15, 2025. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. 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 previously issued a proposed finding that the amendments involve no significant hazards consideration published in the *Federal Register* on February 20, 2024 (89 FR 12873) and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Letter from Florida Power and Light Company, "Turkey Point Nuclear Generating Station, Unit 3 and 4, Docket Nos. 50-250 and 50-251, Renewed Facility Operating Licenses DPR-31 and DPR-41, License Amendment Request 278, Incorporate Advanced Fuel Products, Extend Surveillance Intervals and 10 CFR 50.46 Exemption Request to Facilitate Transition to 24-Month Fuel Cycles," November 15, 2023, ADAMS Accession No. ML23320A028 (Public); ML23320A029 (Proprietary).

2. Letter from Florida Power and Light Company, "Turkey Point Nuclear Generating Stations, Unit 3 and 4, Supplement to License Amendment Request 278, Incorporate Advanced Fuel Products, Extend Surveillance Intervals and 10 CFR 50.46 Exemption Request to Facilitate Transition to 24-Month Fuel Cycles," February 9, 2024, ADAMS Accession No. ML24040A190.

3. Letter from Florida Power and Light Company, "Turkey Point Nuclear Generating Station, Unit 3 and 4 Docket Nos. 50-250 and 50-251 Subsequent Renewed Facility Operating Licenses DPR-31 and DPR-41, Turkey Point Nuclear Plant, Unit 3 and 4, Docket Nos. 50-250 and 50-251, Subsequent Renewed Facility Operating Licenses DPR-31 and DPR-41 Response to Requests for Additional Information Regarding Turkey Point License Amendment Request (278) to Facilitate a Transition to 24-Month Fuel Cycles," October 3, 2024, ADAMS Accession No. ML24278A038 (Public); ML24278A040 (Proprietary).

4. Letter from Florida Power and Light Company, "Response to Request for Additional Information Regarding Turkey Point License Amendment Request (278) to Facilitate a Transition to 24-Month Fuel Cycles," October 31, 2024, ADAMS Accession No. ML24305A144.

5. Letter from Florida Power and Light Company, "Turkey Point Nuclear Generating Station, Unit 3 and 4 Docket Nos. 50-250 and 50-251 Subsequent Renewed Facility Operating Licenses DPR-31 and DPR-41, Turkey Point Nuclear Plant, Unit 3 and 4 Docket Nos. 50-250 and 50-251 Subsequent Renewed Facility Operating Licenses DPR-31 and DPR-41 Response to Requests for Additional Information Regarding Turkey Point License Amendment Request (278) to Facilitate a Transition to 24-Month Fuel Cycles," November 12, 2024, ADAMS Accession No. ML24317A059.

6. 1967/07/11 Atomic Energy Commission Proposed General Design Criterion, 10 CFR Part 50, Appendix A (1967 Proposed GDC), ADAMS Accession No. ML043310029.

7. Generic Letter (GL) 91-04, “Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle,” April 2, 1991, ADAMS Accession No. ML031140501.

8. WCAP-18888-P, “Westinghouse Setpoint Methodology for Protection Systems, Turkey Point Units 3 & 4, 24 Month Fuel Cycle,” Revision 0, ADAMS Accession No. ML24040A191 (Proprietary).

9. Regulatory Guide, 1.105, Revision 4, “Setpoints for Safety-Related Instrumentation,” February 2021, ADAMS Accession No. ML20330A329.

10. Regulatory Issue Summary (RIS) 2006-17, “NRC Staff Position on the Requirements of 10 CFR 50.36, ‘Technical specifications,’ Regarding Limiting Safety System Settings during Periodic Testing and Calibration of Instrument Channels,” August 24, 2006, ADAMS Accession No. ML051810077.

11. TSTF-493, “Clarify Application of Setpoint Methodology for LSSS Functions,” ADAMS Accession No. ML101160026.

12. WCAP-12745-P, “Westinghouse Setpoint Methodology for Protection Systems ‐ Turkey Point Units 3 & 4” (Proprietary).

13. WCAP-17070-P, “Westinghouse Setpoint Methodology for Protection Systems Turkey Point Units 3 & 4 (Power Uprate to 2644 MWt - Core Power),” Revision 1 (Proprietary).

14. Turkey Point Nuclear Plant, Units 3 and 4, “Cover Letter and Amendments- Issuance of Amendments Regarding Extended Power Uprate,” June 15, 2012, ADAMS Accession No. ML11293A365.

15. NUREG-1431, Volume 1, Revision, 5, “Standard Technical Specifications, Westinghouse Plants,” September 2021, ADAMS Accession No. ML21259A155.

16. RG 1.52, “Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants,” Revision 2, March 1978, ADAMS Accession No. ML003740139.

17. NRC Email capture, “Turkey Point Nuclear Generating, Unit Nos. 3 and 4, Request for Additional Information, Transition to 24-Montyh Fuel Cycles (L-2023-0161, September 17, 2024, ADAMS Accession No. ML24261B806.

18. WCAP-17504-NP-A, Revision 1, “Westinghouse Generic Setpoint Methodology,” October 31, 2016, ADAMS Accession No. ML16314A091.

19. WCAP-9272-P-A, Revision 0, “Westinghouse Reload Safety Evaluation Methodology” (Proprietary).

20. WCAP-18546, Revision 0, “Westinghouse AXIOM® Cladding for Use in Pressurized Water Reactor Fuel,” March 2023, ADAMS Accession Package No. ML23089A066, ML23089A006 (Public), ML23089A065 (Proprietary).

21. WCAP-18482, Revision 0, “Westinghouse Advanced Doped Pellet Technology (ADOPTTM) Fuel,” ADAMS Accession Package No. ML22259A187, ML22259A287 (Public), ML22259A186 (Proprietary).

22. WCAP-17642-NP-A, Revision 1, “Westinghouse Performance Analysis and Design Model (PAD5),” November 2017, ADAMS Accession No. ML17338A396 (Public), ML17334A841 (Proprietary).

23. WCAP-11397-P-A, “Revised Thermal Design Procedure,” April 1989 (Proprietary).

24. WCAP-8762-P-A, “New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux In Rod Bundles With Mixing Vane Grids,” July 1984 (Proprietary).

25. WCAP-14565-P-A, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” October 1999, ADAMS Accession No. ML081280711.

26. WCAP-16996, Revision 1, “Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology),” ADAMS Accession Package No. ML16343A238, ML16343A243 (Proprietary).

27. Turkey Point Nuclear Generating Unit Nos. 3 and 4 – Issuance of Amendment Nos. 296 And 289 Regarding Implementation of Full Spectrum Loss-Of-Coolant Accident (FSLOCA) Methodology (EPID L-2021-LLA-0070), March 24, 2022, ADAMS Accession No. ML22028A066.

28. WCAP-7907-P-A, “LOFTRAN Code Description,” April 1984 (Proprietary).

29. WCAP-14882-P-A, “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis,” April 1999 (Proprietary).

30. WCAP-14882-S1-P-A, “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactors Non-LOCA Safety Analyses, Supplement 1 -Thick Metal Mass Heat Transfer Model and NOTRUMP-Based Steam Generator Mass Calculation Method,” October 2005 (Proprietary).

31. WCAP-7979-P-A, “TWINKLE – A Multi-Dimensional Neutron Kinetics Code,” January 1975 (Proprietary).

32. WCAP-7908-A, “FACTRAN – A Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod,” December 1989, ADAMS Accession No. ML20012C559.

33. WCAP-7588, Revision 1-A, “An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors using Spatial Kinetics Methods,” January 1975 (Proprietary).

34. Turkey Point Units 3 and 4 Extended Power Uprate Licensing Report Attachment 4 Appendix A Safety Evaluation Report Compliance, ADAMS Accession No. ML103560178.

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8.0 ABBREVIATIONS

Abbreviation	Definition
AC	Alternating Current
ADV	Atmospheric Dump Values
ADOPT	Advanced Doped Pellet Technology
AEC	Atomic Energy Commission
AFT	As-Found Tolerance
AFW	Auxiliary Feedwater
AFWS	Auxiliary Feedwater System
AH	Ampere-hour
ALT	As-Left Tolerance
ANS	American Nuclear Society
ANSI	American National Standards Institute

Abbreviation	Definition
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
AV	Allowable Value
B&PV	Boiler and Pressure Vessel
BOC	Beginning-of-Cycle
CAP	Containment Accident Pressure
CAP	Corrective Action Program
CCW	Component Cooling Water
CE	Combustion Engineering
CFR	*Code of Federal Regulations*
CLOF	Complete Loss of Flow
COLR	Core Operating Limit Report
COT	Channel Operational Test
CP	Cathcart-Pawel
Cr	Chromium
CRE	Control Rod Ejection
CREVS	Control Room Emergency Ventilation System
CSS	Containment Spray System
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control System
CWO	Core Wide Oxidation
DBA	Design Basis Accident
DC	Direct Current
DEG	Double-Ended Guillotine
DER	Double-Ended Rupture
DFBN	Debris Filter Bottom Nozzle
DEHL	Double-Ended Hot Leg
DEPS	Double-Ended Pump Suction
DLM	Diffusion Layer Model
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
Δp	= $\Delta k/k$ or Reactivity
ECCS	Emergency Core Cooling System
ECR	Equivalent Clad Reacted
EDG	Emergency Diesel Generator
EFPY	Effective Full-Power Year
EM	Evaluation Model
EOC	End-of-Cycle
EPU	Extended Power Uprate
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
^{o}F	Degrees Fahrenheit
FA	Fuel Assembly
FSLOCA	Full Spectrum LOCA
FCEP	Fuel Criteria Evaluation Process
FCM	Fuel Centerline Melt

Abbreviation	Definition
FPL	Florida Power and Light Company
FWLB	Feedwater Line Break
GAD	Gadolinia or Gad
GD_2O_3	Gadolinia
GDC	General Design Criteria
GL	Generic Letter
gpm	gallons per minute
GTRF	Grid to Rod Fretting Wear
GWd	Gigawatt days
HELB	High Energy Line Break
HFP	Hot Full Power
HHSI	High Head Safety Injection
HPU	Hydrogen Pickup
HVFD	Hafnium Vessel Flux Depression
HZP	Hot Zero Power
ICU	Initial Condition Uncertainties
ICW	Intake Canal Water
IFBA	Integral Fuel Burnable Absorber
IFM	Intermediate Flow Mixer
L&C	Limitation and Condition
LAR	License Amendment Request
LBLOCA	Large Break Loss-of-Coolant Accident
LCO	Limiting Condition of Operation
LHGR	Linear Heat Generation Rate
LOAC	Loss of Non-Emergency AC Power
LOCA	Loss-of-Coolant Accident
LOL/TT	Loss of Load/Turbine Trip
LONF	Loss of Normal Feedwater Flow
LOOP	Loss-of-Offsite Power
LSSS	Limiting Safety System Setting
LWR	Light Water Reactor
M&E	Mass and Energy
MEM	Minimum ECR Margin
MDLM	Mist Diffusion Layer Model
MLO	Maximum Local Oxidation
MSS	Main Steam System
MSCV	Main Steamline Chack Valve
MSIV	Main Steam Isolation Valve
MSSV	Main Steam System Valve
MTO	Thermal-Hydraulic Margin to Overfill
MTC	Moderator Temperature Coefficient
MTU	Metric Ton Uranium
MWt	Megawatt thermal
NPSH	Net Positive Suction Head
NPSHA	Available NPSH
NPSHR	Required NPSH
NRC	Nuclear Regulatory Commission

Abbreviation	Definition
NTS	Nominal Trip Setpoint
NTSP	Nominal Trip Setpoint
OMS	Overpressure Mitigating System
OPA	Offsite power available
PAM	Post Accident Monitoring
pcm	per cent mille (one-thousandth of a percent)
PCMI	Pellet-Cladding Mechanical Interaction
PCT	Peak Cladding Temperature
PIV	Pressure Isolation Valve
PLHGR	Peak Linear Heat Generation Rate
PLOF	Partial Loss of Flow
PORV	Power Operated Relief Valve
ppm	parts per million
pRXA	Partially RXA
psi	pounds per square inch
psig	pounds per square inch gauge
PSV	Pressurizer Safety Valve
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCA	Rack Calibration Accuracy
RCC	Rod Cluster Control
RCCA	RCC Assembly
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RIP	Rod Internal Pressure
RIS	Regulatory Issue Summary
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RTDP	Revised Thermal Design Procedure
RTN	Reconstitutable Top Nozzle
RTP	Rated Thermal Power
RTS	Reactor Trip System
RWFS	RCCA Withdrawal from Subcritical
RXA	Recrystallized Annealed
SAL	Safety Analysis Limit
SBLOCA	Small Break Loss-of-Coolant Accident
SDM	Shutdown Margin
SE	Safety Evaluation
SER	Safety Evaluation Report
SFA	Surveillance Failure Analysis
SFCP	Surveillance Frequency Control Program
SG	Steam Generator
SGTP	Steam Generator Tube Plugging
SGTR	Steam Generator Tube Rupture

Abbreviation	Definition
SI	Safety Injection
SL	Safety Limit
SLB	Steamline Break
SLRA	Subsequent License Renewal Application
Sn	Discrete Coordinates
SR	Surveillance Requirement
SRP	Standard Review Plan
SRA	Stress-Relieved Annealed
SRSS	Square root sum of the squares
TCD	Thermal Conductivity Degradation
T-H	Thermal-Hydraulic
TLU	Total Loop Uncertainty
TR	Topical Report
TRS	Technical Requirememt Surveillance
TS	Technical Specification
TSTF	Technical Specification Task Force
UF	Under-Frequency
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
UO_2	Uranium di-Oxide
µg/gU	Microgram/Gram Uranium
µm	Micrometer
UV	Under Voltage
VFTP	Ventilation Filter Testing Program
WABA	Wet Annular Burnable Absorbers
WCAP	Westinghouse Commercial Atomic Program
WO	Work Order

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SUBJECT: TURKEY POINT NUCLEAR GENERATING, UNIT NOS. 3 AND 4 – ISSUANCE OF AMENDMENT NOS. 301 AND 294 REGARDING INCORPORATION OF ADVANCED FUEL PRODUCTS AND EXTENSION OF SURVEILLANCE INTERVALS TO FACILITATE TRANSITION TO A 24-MONTH FUEL CYCLE (EPID L-2023-LLA-0161) DATED FEBRUARY 12, 2025.

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ML25021A189 (Package)
ML25021A184 (Proprietary)
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