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

April 3, 2018

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

SUBJECT: TURKEY POINT NUCLEAR GENERATING UNIT NO. 3 – ISSUANCE OF AMENDMENT REGARDING THE ONE-TIME EXTENSION OF THE CONTAINMENT SPRAY SYSTEM COMPLETION TIME (EPID L-2017-LLA-0423)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 280 to Renewed Facility Operating License No. DPR-31 for Turkey Point Nuclear Generating Unit No. 3. The amendment changes the Technical Specifications in response to the application from Florida Power & Light Company dated December 18, 2017 (L-2017-213), as supplemented by letter L-2018-056 dated February 16, 2018.

The amendment revises the Technical Specifications to extend the completion time for the Containment Spray system from 72 hours to 14 days to allow for the repair of the 3A Containment Spray pump. The extension is applicable on a one-time basis during Unit 3 Cycle 29.

The NRC staff's safety evaluation of the amendment is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

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

Docket No. 50-250

Enclosures:
1. Amendment No. 280 to DPR-31
2. Safety Evaluation

cc: Listserv

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 280
Renewed License No. DPR-31

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

   A. The application for amendment by Florida Power & Light Company (the licensee) dated December 18, 2017, as supplemented by letter dated February 16, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Renewed Facility Operating License and Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Renewed Facility Operating License No. DPR-31 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 280, are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Brian W. Tindell

Brian W. Tindell, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: April 3, 2018

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 280 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-31

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-250

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

Replace the following page of the Appendix A Technical Specifications with the attached page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3/4 6-12	3/4 6-12

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

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

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

A. Maximum Power Level

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

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 280, are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Final Safety Analysis Report

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

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

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and manually transferring suction to the containment sump via the RHR System.

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

ACTION:

a. With one Containment Spray System inoperable restore the inoperable Spray System to OPERABLE status within 72 hours** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two Containment Spray Systems inoperable restore at least one Spray System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both Spray Systems to OPERABLE status within 72 hours of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position* and that power is available to flow path components that require power for operation;

b. By verifying that on recirculation flow, each pump develops the indicated differential pressure, when tested in accordance with the INSERVICE TESTING PROGRAM.

   Containment Spray Pump $\geq$241.6 psid while aligned in recirculation mode.

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

---

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

** During Unit 3 Cycle 29 only, a one-time extension from 72 hours to 14 days is allowed to perform 3A Containment Spray Pump (3P214A) planned maintenance, provided the following compensatory measures are in place:

- 3B Containment Spray Pump and associated electrical breaker [Guarded]
- 3A, 3B and 3C Emergency Containment Coolers and associated electrical breakers [Guarded]
- 3A, 3B Emergency Diesel Generators [Guarded]
- Unit 3 Startup Transformer and associated onsite AC power distribution system [Guarded]

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 280 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-31

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-250

1.0 INTRODUCTION

By application dated December 18, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17353A492), as supplemented by letter dated February 16, 2018 (ADAMS Accession No. ML18050A001), Florida Power & Light Company (the licensee or FPL) requested changes to the Technical Specifications (TSs) for Turkey Point Nuclear Generating Unit No. 3 (Turkey Point), which are contained in Appendix A of Renewed Facility Operating License No. DPR-31. The licensee proposed to revise TS limiting condition for operation (LCO) 3.6.2.1, "Containment Spray System," ACTION a, to extend the completion time for restoring an inoperable Containment Spray (CS) system from 72 hours to 14 days to allow for the repair of the 3A CS pump. The proposed extension would be applicable on a one-time basis during Unit 3 Cycle 29.

By electronic mail (e-mail) dated February 5, 2018 (ADAMS Accession No. ML18037A150), the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff sent the licensee a request for additional information (RAI). The licensee responded to the RAI by letter dated February 16, 2018. The licensee's letter provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the original proposed no significant hazards consideration (NSHC) determination that was published in the *Federal Register* (FR) on January 30, 2018 (83 FR 4285).

2.0 REGULATORY EVALUATION

2.1. CS System

The CS system provides containment heat removal capability by spraying cool, borated water into the Containment atmosphere in the event of a loss-of-coolant accident (LOCA) or a main steam line break (MSLB). The CS and Emergency Containment Cooling (ECC) systems are designed to prevent Containment pressure from exceeding design limits. In the event of a LOCA or MSLB, the minimum available Containment Spray and ECC system equipment shall maintain the Containment pressure and temperature below the containment structural design values consistent with the limiting single active failure. The CS system is sized to supply the necessary post-accident cooling capacity to reduce Containment pressure following blowdown and cooling of the core by safety injection for all pipe-break sizes up to, and including, a hypothetical instantaneous circumferential-rupture of the largest reactor coolant pipe. The CS

system is also credited for post-accident Containment atmosphere iodine removal and hydrogen mixing.

The CS system consists of two CS pumps, spray ring headers and nozzles, and associated piping and valves. All CS system components, piping, structures, and power supplies are designed to seismic Class I criteria. The CS pumps, which take suction directly from the Refueling Water Storage Tank, are located in the Auxiliary Building and are protected by concrete enclosures designed to withstand missile impact. The CS system also utilizes the two Residual Heat Removal (RHR) pumps, two RHR heat exchangers, and associated valves and piping of the Safety Injection system for long term recirculation phase Containment cooling. The current Containment analysis models CS flow as a function of Containment pressure. For the limiting case of the Containment analysis, Containment Spray System flow capability at 50 pounds per square inch gauge is 1293 gallons per minute during the initial injection phase and 1575 gallons per minute during the cold leg recirculation phase.

Containment spray is actuated on coincident signals of two-out-of-three high and two-out-of-three high-high containment pressure signal channels. The system can also be manually actuated in the Control Room.

TS LCO 3.6.2.1 requires that two independent CS systems be OPERABLE and that each system is capable of taking suction from the Refueling Water Storage Tank and manually transferring suction to the containment sump, via the RHR system. In the event of an inoperable CS system, ACTION a requires the inoperable CS system to be restored within 72 hours. If the inoperable system cannot be restored within 72 hours, then ACTION a requires the plant to be shut down within the following 6 hours.

2.2 Licensee's Proposed Changes

The licensee proposed to revise TS LCO 3.6.2.1, ACTION a, as follows (additional text shown underlined):

ACTION:

a. With one Containment Spray System inoperable restore the inoperable Spray System to OPERABLE status within 72 hours** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

** During Unit 3 Cycle 29 only, a one-time extension from 72 hours to 14 days is allowed to perform 3A Containment Spray Pump (3P214A) planned maintenance, provided the following compensatory measures are in place:

- 3B Containment Spray Pump and associated electrical breaker [Guarded]
- 3A, 3B and 3C Emergency Containment Coolers and associated electrical breakers [Guarded]
- 3A, 3B Emergency Diesel Generators [Guarded]
- Unit 3 Startup Transformer and associated onsite AC power distribution system [Guarded]

The licensee stated that the purpose of the proposed changes is to allow the licensee to perform an at-power modification of the 3A CS pump skid and foundation assembly. Based on

similar modifications performed on the 3B and Unit 4 CS pumps, the licensee estimates that the modification to the 3A CS pump will require approximately 10 days to implement.

2.3 Regulatory Review

The NRC staff reviewed the licensee's application to determine whether (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 the activities proposed 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 the health and safety of the public. The NRC staff considered the following regulatory requirements, guidance, and licensing and design-basis information during its review of the proposed changes.

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Paragraph 50.36(a)(1) states, in part, that each applicant for an operating license shall include in the application proposed TSs in accordance with the requirements of 10 CFR 50.36, "Technical specifications." Paragraph 50.36(c) of 10 CFR requires that the TSs include items in the following categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) surveillance requirements; (4) design features; and (5) administrative controls. Paragraph 50.36(c)(2) states, in part, that LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility and that when an LCO is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.

The Turkey Point 3 and 4 design approval for its construction phase was based on the proposed general design criteria (GDC) published by the Atomic Energy Commission in the *Federal Register* on July 11, 1967 (32 FR 10213). Section 1.3, "General Design Criteria," of the Turkey Point Updated Final Safety Analysis Report (UFSAR) (ADAMS Accession No. ML16330A180) describes the Turkey Point GDC and lists other sections of the UFSAR that describe the GDC in more detail. As it relates to this request, the NRC staff considered Turkey Point 3 and 4 GDC 52, which states, "Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure, this system shall perform its required function, assuming failure of any single active component."

NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 16.1, Revision 1, "Risk-Informed Decision making: Technical Specifications" (ADAMS Accession No. ML070380228), states that licensees submitting risk information should address each of the principles of risk-informed regulation addressed in Regulatory Guide (RG) 1.177 "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications."

RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (ADAMS Accession No. ML17317A256), describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed licensing-basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

RG 1.177, Revision 1 (ADAMS Accession No. ML100910008), describes an acceptable risk-informed approach to TS changes, specifically for changes to Completion Times (CTs). This RG also provides risk acceptance guidelines for evaluating the results of such assessments and lists five key principles that TS changes should meet.

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (ADAMS Accession No. ML090410014), describes an acceptable approach for determining whether the quality of the probabilistic risk assessment (PRA) models, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA models can be used in regulatory decision making for light-water reactors.

Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation" (ADAMS Accession No. ML070650428), describes how the NRC will implement its technical adequacy review of plant-specific PRAs used to support risk-informed licensing actions after the issuance of RG 1.200.

"Safety Goals for the Operations of Nuclear Power Plants" (Commission's Safety Goal Policy), 51 FR 28044 (August 4, 1986), as corrected and republished at 51 FR 30028 (August 21, 1986), establishes the Commission's goals that broadly define an acceptable level of radiological risk.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's application to determine if the proposed changes are consistent with the guidance, regulations, and plant-specific design and licensing basis information discussed in Section 2.3 of this safety evaluation (SE).

3.1 Method of Staff Review

An acceptable approach for making risk-informed decisions about proposed TS changes, including both permanent and temporary changes, is to show that the proposed changes meet the five key principles stated in RG 1.174, Section 2, and RG 1.177, Section B. These key principles are:

Principle (1): The proposed change meets the current regulations unless it is explicitly related to a requested exemption.

Principle (2): The proposed change is consistent with the defense-in-depth philosophy.

Principle (3): The proposed change maintains sufficient safety margins.

Principle (4): When the proposed change results in an increase in core damage frequency (CDF) and/or large early release frequency (LERF), the increase should be small and consistent with the intent of the Commission's Safety Goal Policy statement.

Principle (5): The impact of the proposed change should be monitored by using performance measures strategies.

The information used in the NRC staff's review of the licensee's risk evaluation is contained in Section 3.0 of the licensee's request dated December 18, 2017, as supplemented by the licensee's letter dated February 16, 2018. To support the assessment of the licensee's PRA capability for use in the risk evaluation, the NRC staff also reviewed the SE in support of the NRC staff's approval of an alternative (Relief Request (RR) No. 4) to the examination

requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI," dated October 26, 2016 (ADAMS Accession No. ML16293A778).

3.2 Key Principle 1: The Proposed Change Meets Current Regulations

The regulation, 10 CFR 50.36(c), specifies the requirements for TSs. The licensee's proposed one-time change to TS LCO 3.6.2.1, ACTION a, to increase the CT for an inoperable CS system affects the maximum allowed time to have only one CS train OPERABLE without shutting down the reactor. The licensee's request does not deviate from the requirements to comply with this regulation nor any other regulation. Therefore the staff concludes that the licensee's proposed change meets existing regulations.

3.3 Key Principle 2: The Proposed Change is Consistent with Defense-in-Depth Philosophy

SRP Chapter 16.1 states that consistency with defense-in-depth philosophy is maintained if the following seven items evaluated below are met:

1. A reasonable balance among prevention of core damage, prevention of containment failure and consequence mitigation is preserved.

Prevention of core damage depends on the ability to continuously remove decay heat after an initiating event or design basis accident (DBA).

The licensee proposes extending the allowed CT of one CS train when the 3A CS pump is removed from service for repair. Because the CS system is not designed to remove core decay heat, the proposed change will not affect the ability of plant systems to remove decay heat from the core; therefore it will have no effect on the prevention of core damage.

The CS system is used to prevent containment pressure from exceeding containment design pressure after a design basis accident (DBA) such as a circumferential rupture of a reactor coolant pipe causing a LOCA or a main steam pipe rupture causing an MSLB accident. The CS system has redundant trains, each of which is capable of performing the systems design function. During the extended CT, one CS train remains operable and is capable of performing the system design function. Additionally the ECC System, which operates in conjunction with the CS system to ensure containment pressure does not exceed its design pressure, remains operable. Further, two trains of Safety Injection (SI) are operable to assist in mitigation of a LOCA. The SI system not only keeps the core covered, but also keeps the water in the core subcooled after a LOCA, making the assumptions in the design basis of the CS system very conservative. This is because the CS system design basis conservatively assumes all core residual heat is released to the containment as steam; however, the SI system absorbs some of this residual heat. As such, the SI system assists in reducing containment pressure after a LOCA. Therefore, the ability to maintain containment pressure below design pressure after a LOCA or MSLB is preserved.

Consequence mitigation considers the radioactivity released outside containment in event of a DBA and the corresponding effect on the public health and safety. By maintaining the capability to keep containment pressure below design limits during a LOCA, the proposed change has a minimal effect on consequence mitigation.

Therefore, the NRC staff considers that there is a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation.

2. Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.

Programmatic activities proposed by the licensee to accomplish the maintenance activity include pre-job briefs, equipment walk downs, and protecting vital plant equipment. The NRC staff finds that these activities do not constitute an over-reliance on programmatic activities, because the planned activities are in addition to the other defense-in-depth attributes discussed in this evaluation and are not the sole method of achieving defense-in-depth.

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

System redundancy, independence, and diversity are normally achieved with two CS trains for Unit 3. During the repair of the 3A CS pump, the 3B CS train will be the only OPERABLE CS train. Regarding the CS system's containment pressure-reduction function, the 3B CS train with two ECC system cooling units can mitigate all DBAs, provided there are no failures in the 3B CS train. The CS system works in conjunction with the ECC system, which will have all three ECC units operable in accordance with TS 3.6.2.2. Two trains of the SI system will also be available during the extended CT. The SI system would keep the water in the core subcooled and thereby reduce the steam released to containment during a LOCA. This adds significant conservatism to the design basis of the CS system, which assumes all core residual heat is released to the containment as steam. The CS and ECC systems will be protected by plant procedures. Regarding the CS system's post-accident iodine removal function, sodium tetraborate decahydrate (NaTB) located in baskets inside containment ensures that post-LOCA sump water pH is maintained between 7 and 8.048 during long-term recirculation and that evolution of iodine will be minimized during the one-time extension. Regarding the CS system's post-accident hydrogen mixing function, ECC system fans also work to disperse local hydrogen concentrations. Further, internal containment structures are designed to prevent hydrogen pocketing. Therefore, the NRC staff finds that system redundancy, independence and diversity is maintained commensurate with the risk associated with a one-time extension of the CT of an inoperable CS system train.

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

In its December 18, 2017, request, the licensee stated:

> Defenses against potential common-cause failures (CCFs) will be maintained by limiting non-essential maintenance and operation of SSCs [structures, systems, and components] having mitigatory roles credited in accident analyses. This includes SSCs providing similar and/or support functions for preventing Containment pressure from exceeding design limits in the unlikely event of a LOCA or MSLB. FPL's risk analysis quantified the possible CCF combinations for the CS pumps and concluded that the risk impact for the proposed Completion Time extension is insignificant. Moreover, FPL has a high degree of confidence that no new CCF mechanisms can be introduced given that the planned 3A CS Pump modification has been successfully applied to the 3B CS Pump and is similar to the 4A and 4B CS Pumps. Modification acceptance testing will confirm 3A CS Pump readiness to return to service.

Limiting non-essential maintenance and operation of SSCs to mitigate DBAs is an appropriate defense against CCF, because it reduces the likelihood of equipment failure during the planned modification of the 3A CS pump. Further, the potential for introduction of a new CCF mechanism has been addressed because the planned modification has been previously performed successfully on the 3B and Unit 4 CS pumps. Therefore, the NRC staff finds that licensee has appropriately addressed CCF.

5. Independence of physical barriers is not degraded

Physical barriers refer to the reactor coolant pressure boundary, the fuel cladding, and the containment. The one-time extension request does not propose changes that would alter the ability to remove decay heat or challenge the reactor coolant pressure boundary; therefore the fuel cladding and reactor coolant pressure boundary are not affected. As previously discussed the proposed change keeps in place the ability to maintain containment pressure below design pressure after a LOCA or MSLB. Therefore, the NRC staff finds that the independence of barriers are not degraded by the proposed one-time extension.

6. Defense against human errors are maintained

In its December 18, 2017, request, the licensee stated:

> Prior to the start and during each shift of the proposed Completion Time extension, a pre-job briefing will be conducted to reinforce expected human performance behaviors and bolster defense-in-depth barriers to human errors. In order to minimize plant challenges, Operators and maintenance shift crews will be briefed on procedures for implementing and maintaining the equipment lineup necessary to perform the planned 3A CSP [containment spray pump] modification. Risk aspects of the proposed Completion Time extension will be emphasized during these briefings. Operators will be additionally briefed on responding to unintended and unforeseen circumstances that may rely on CS system operability during the proposed Completion Time extension.

The NRC staff finds that defense against human errors will be maintained because the licensee's use of pre-job briefs make it less likely that errors will be committed during the planned maintenance, or that other plant staff will perform an incorrect action.

7. The intent of plant design is maintained

Turkey Point 3 and 4 GDC 52, states:

> Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure, this system shall perform its required function, assuming failure of any single active component.

Turkey Point is designed with two trains of CS. Only one train will be available for the extended CT of 14 days proposed by the licensee, but the intent of plant design as stated above is not being permanently changed. For a specified one-time 14 day period, the single-failure criterion for this system is relaxed. With the 3A CS pump and train inoperable, the 3B CS train can mitigate all design basis accidents and events, thus the safety function is maintained. Additionally, the need for the CS system is unlikely, because a LOCA or MSLB is highly improbable during the extended CT. Also, two SI trains and three ECC system coolers would be available to reduce containment pressure if a LOCA did occur. Further, the 3B CS train and

3 ECC system coolers would be available to reduce containment pressure if a MSLB were to occur. Therefore the NRC staff finds that the intent of plant design is maintained.

3.4 Key Principle 3: The Proposed Change Maintains Sufficient Safety Margins

The extended CT is not in conflict with Codes and Standards approved for use by the NRC relevant to the CS system and associated supported systems. Safety analysis acceptance criteria as specified in the UFSAR, particularly for large-break LOCA, are met during the extended CT, provided that the 3B CS train does not fail. As discussed in Section 3.5.1.2 of this SE, the probability of a LOCA or MSLB and failure of the 3B CS train during the 14 day outage is highly unlikely and within the core damage frequency criteria approved by the NRC. Therefore, the NRC staff considers the safety margins are maintained for this one-time CT extension.

3.5 Key Principle 4: Change in Risk Is Consistent With the Safety Goal Policy Statement

The evaluation presented below addresses the NRC staff's philosophy of risk-informed decision making. For proposed changes resulting in a change in CDF or risk, the increase should be small and consistent with the intent of the Commission's Safety Goal Policy Statement. The licensee stated that the Turkey Point PRA was utilized to evaluate the impact on CDF and LERF to support requesting an extension to the CT to allow only one operable CS pump from 72 hours to 14 days. The CS pumps are safety-related and therefore explicitly modeled in the average maintenance model. The NRC staff evaluated Key Principle 4 using the three-tiered approach described in SRP Chapter 16.1 and RG 1.177:

- Tier 1 - The first tier evaluates the licensee's PRA and the impact of the change on plant operational risk, as expressed by the change in CDF and change in LERF. The change in risk is compared to the acceptance guidelines presented in RG 1.174. The first tier also aims to ensure that plant risk does not increase unacceptably during the period when equipment is taken out of service per the license amendment, as expressed by the incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). The incremental risk is also compared to the acceptance guidelines presented in RG 1.177.

- Tier 2 - The second tier addresses the need to preclude potentially high-risk plant configurations that could result if equipment, in addition to that associated with the proposed license amendment, are taken out of service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The objective of this part of the review is to ensure that appropriate restrictions on dominant risk-significant plant configurations associated with the CT extension are in place.

- Tier 3 - The third tier addresses the licensee's overall configuration risk management program (CRMP). The purpose of the CRMP is to ensure that equipment removed from service prior to or during the proposed extended CT period will be appropriately assessed from a risk perspective.

This three-tiered approach ensures that adequate programs and procedures are in place to identify risk-significant plant configurations resulting from maintenance or other operational activities and to take appropriate compensatory measures to avoid such configurations. In order to determine whether the PRA used in support of the proposed CT extension is of sufficient quality, scope, and level of detail, the NRC staff evaluated the relevant information

provided by the licensee in its submittal, as supplemented, and considered the findings of recent PRA reviews. The NRC staff's review of the licensee's submittal focused on the ability of the licensee's PRA model to analyze the risks stemming from the proposed CT extension and did not involve an in-depth review of the licensee's PRA.

3.5.1 Tier 1: PRA Capability and Insights

3.5.1.1 Evaluation of PRA Acceptability

Section 2.3.2 of RG 1.177 states that, as a minimum, the licensee should perform evaluations of CDF and LERF to support any risk-informed changes to TS. The scope of the analysis should include all hazard groups (i.e., internal events (IE), internal flooding (IF), internal fires, seismic events, high winds, transportation events, and other external hazards). Section 2.3.1 of RG 1.174 states that a qualitative treatment of the missing modes and hazard groups may be sufficient when the licensee can demonstrate that those risk contributions would not affect the decision.

RG 1.200 describes one acceptable approach for determining whether the technical acceptability of a PRA is sufficient for use in regulatory decision making for light-water reactors. The purpose of RG 1.200 is: a) to provide guidance to licensees for use in determining the technical acceptability of the base PRA used in a risk-informed regulatory activity, and b) to endorse industry standards and peer-review guidance.

As discussed below for the evaluation of the licensee's proposed change, the scope of the licensee's evaluation should include an assessment of the change in risk for IE, IF, internal fires, and seismic events. The licensee provided a quantitative assessment of the change in risk using a PRA model, which includes an assessment of IE and IF. For internal fires and seismic hazards, the licensee provided a qualitative assessment of the change in risk that is based on insights gained from the IE and flooding PRA model.

3.5.1.1.2 Internal Events and Internal Flooding

The licensee's PRA model has been updated several times since the Individual Plant Examination to keep it consistent with the as-built/as-operated plant, to incorporate new and revised plant-specific thermal hydraulic results, and to incorporate new and revised PRA methodologies. The PRA model and results are maintained as controlled documents, and for this evaluation, the licensee used the same PRA model that was used to support RR No. 4, which was approved in 2016. This model is an at-power Level I that includes IE and IF.

Attachment A of RR No. 4 (ADAMS Accession No. ML16033A355) describes two focused-scope peer reviews of the licensee's PRA model. A focused-scope peer review was performed for the IE and IF portions of the PRA model in April 2011. This peer review assessed the human reliability analysis (HRA) elements of the standard. Another focused-peer review was performed in October 2013 to assess upgrades to the PRA model in areas of CCF analysis (level 2) and interfacing system LOCAs (ISLOCAs). Both focused-scope peer reviews were performed using ASME/American Nuclear Society (ANS) Standard, RA-Sa-2009, "Addenda to ASME RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," and RG 1.200, Revision 2. Tables 1, 2, and 3 of Attachment A to RR No. 4 provided lists of the facts and observations (F&Os) from the 2002 peer reviews, as well as the 2011 and 2013 focused-scope peer reviews. In RR No. 4, the licensee indicated that the F&Os from the 2013 peer review had not yet been resolved in its PRA model, which had also been reviewed by the NRC staff and found acceptable, as

described in the July 2015 SE supporting Amendment Nos. 263 and 258 (ADAMS Accession No. ML15166A320) regarding the relocation of specific surveillance frequency requirements to a licensee controlled program.

In its letter dated July 27, 2016 (ADAMS Accession No. ML16217A458), in response to RAIs regarding RR No. 4, the licensee provided additional information on the impact and the resolution of the remaining 2013 peer-reviewed F&Os. Finding DA-05-01 stated that the global common cause event needs to account for the common cause combinations not included explicitly, and for several 6-component groups, the 5-of-6 term was not included, and the 6-of-6 term was not adjusted in the licensee's PRA. The licensee stated that, "the common cause alpha factors were updated and the CAFTA [cutset and fault tree analysis] CCF tool was used to resolve this F&O." In Table 3 of RR No. 4, the supporting requirement associated with finding DA-06-01 was determined to not be met because the CCF notebook did not include a review of plant failure data for common cause events. For DA-06-01, the licensee stated that the issue was addressed in a data update and that no plant specific CCFs were found.

In Table 1, "IE and IF PRA Peer Review – Findings and Resolutions," submitted with RR No. 4, the licensee provided two F&Os, LE-G5 and IE-C14 that the peer reviews identified were not met. Finding LE-G5-01 stated, in part, that conservative treatment of some phenomenon can affect LERF quantification, which in turn impacts LERF and delta LERF results when applying RG 1.174 guidelines in risk-informed changes to the licensing basis. For SR IE-C14, the peer review team identified several findings that determined the PRA model was not adequate to support modeling steam generator tube rupture (SGTR) and ISLOCA scenarios.

In its e-mail dated February 5, 2018, in APLA RAI 01.b.1, the NRC staff requested the licensee to provide additional information to identify any new F&Os generated since approval of RR No. 4 and identify any open IE and IF F&Os that might affect the CDF and LERF estimates where the delta CDF and delta LERF values for the requested CT extension would no longer meet the RG 1.177 and RG 1.174 risk metrics. In its response to APLA RAI 01.b.1 dated February 16, 2018, the licensee indicated that the CS system is modeled in its PRA model only as an environmental qualification support system for the RHR suction valves (MOV-3-750 and MOV-3-751) and the containment sump level instrumentation. The licensee stated that because the CS system only affects these components, its impact on overall risk is minimal. To further address CS system impact on risk as modeled in the PRA, the licensee stated in its response to APLA RAI 01.b.2 that open IE and IF F&Os were reviewed and none had been identified to have a significant effect on the risk impact of the one-time extension for the CS system train out-of-service. In the licensee's review of open findings specifically related to LERF (i.e., IE-C14-01, IE-C14-02, IE-C14-03, IE-C14-04, and IE-C14-05), the licensee stated that a quantitative analysis using a PRA model update has found the impact to be minimal. The licensee indicated that impact to risk does not result in any LERF increase and is only with respect to the baseline LERF, consequently the RG 1.177 risk metrics for delta CDF and delta LERF will be met.

The NRC staff recognizes that the type of containment (i.e., large, dry containment) typically has a small conditional probability of early structural failure and that the LERF is typically dominated by events that result in a release that bypasses the containment, such as SGTRs and ISLOCAs. However, the unavailability of a train of the CS system can impact the probability of early containment failure since it supports the containment heat removal function, in conjunction with the ECC system (fan coolers). The NRC staff also considered information from NUREG/CR-6595, Revision 1, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events" (ADAMS Accession No. ML043240040), and NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant

Performance" (ADAMS Accession No. ML063470259). Based on the above considerations, the NRC staff concludes that the one-time extension of the CT for the repair of the 3A CS pump is not a significant contributor to CDF and LERF and that the licensee's IE and IF PRA is acceptable to use in support of proposed change.

3.5.1.1.3 Internal Fire

In support of RR No. 4, for the development of the fire PRA (FPRA), the licensee modified its IE PRA model to capture the effects of fire. The licensee evaluated the technical adequacy of the FPRA model by conducting a peer review using the guidance provided in Nuclear Energy Institute (NEI) 07-12, Revision 1, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines" (ADAMS Accession No. ML102230070), and the FPRA part (Part 4) of ASME/ANS-RA-Sa-2009, as clarified by RG 1.200, Revision 2. The licensee also conducted a follow-on focus-scoped peer review to address the fire scenario selection and analysis, HRA, and plant response model technical elements. The full-scope peer review of the FPRA was performed in February 2010, and the follow-on focused scope peer review was performed in March 2012.

In section 3.3.1.4 of its request dated December 18, 2017, the licenses stated that "the Turkey Point fire model contains various components that do not have their associated cables mapped and were therefore conservatively assumed to fail. The CS pumps were treated this way and therefore their assumed failure as part of this risk evaluation had no impact." In addition to the quantitative risk aspects, the NRC staff also considered the traditional evaluation discussed in SE Sections 3.2 and 3.4 above, and the licensee's proposed risk management actions (RMAs) discussed in SE Section 3.5.1.1.5 below. The NRC staff concludes that the CS system is not a significant contributor to CDF and LERF because the failure in the risk evaluation resulted in no impact and defense-in-depth will be maintained during the proposed extension. Therefore, the NRC staff finds that the treatment of the CS pump in the FPRA is reasonable for the one-time extension of the TS CT for the repair of the CS 3A pump.

3.5.1.1.4 Seismic and Other External Hazards

Ideally, quantitative evaluations should be made; however, qualitative arguments, bounding analyses, and compensatory measures may also be used. The licensee stated that the impact of external events on the increase in risk associated with this proposed change was not explicitly calculated due to a lack of seismic and other external events PRA models, which are not necessarily realistic and suitable for configuration-specific analysis. For consideration of seismic and other external events, the licensee assumed the contribution to be equal to the IE contribution.

In addition to the quantitative risk aspects, the NRC staff also considered the traditional evaluation discussed in SE Sections 3.2 and 3.4 above, and the licensee's proposed RMAs discussed in SE Section 3.5.1.1.5 below. As discussed above, and demonstrated by the very small change in risk values, the CS system's only impact to CDF through improving the environment inside containment for some valves, has effectively no impact on LERF. The NRC staff concludes that the one-time extension of the CT for the repair of the 3A CS pump is not an important contributor to CDF and LERF and that an increase in contribution by a factor of two for seismic and other external events is reasonable for the one-time extension of the CT for the repair of the CS 3A pump. The total risk from these other external events is generally much less than for IE and fires; therefore, the NRC staff finds that doubling the IE risk contribution is a bounding estimate.

3.5.1.1.5 PRA Acceptability Conclusions

RG 1.177 states that "the licensee should provide the rationale that supports the acceptability of the proposed changes by integrating probabilistic insights with traditional considerations to arrive at a final determination of risk." The NRC staff reviewed the RMAs discussed in the licensee's request dated December 18, 2017, which the licensee stated would be implemented for the duration of the proposed one-time 3A CS pump CT extension. The licensee stated that "by creating these multiple independent and redundant layers of defense, compliance with applicable general design criteria, national standards, and engineering principles and the integrity of barriers to core damage and containment failure are assured. The RMAs identified by the licensee include the following:

- 3B CS Pump and associated electrical breaker [Guarded]
- 3A, 3B and 3C Emergency Containment Coolers and associated electrical breakers [Guarded]
- 3A, 3B Emergency Diesel Generators [Guarded]
- Unit 3 Startup Transformer and associated onsite AC power distribution system [Guarded]
- Prior to the start and during each shift of the proposed Completion Time extension, a pre-job briefing will be conducted to:
  1. Reinforce expected human performance behaviors and bolster defense-in-depth barriers to human errors.
  2. Operators and maintenance shift crews will be briefed on procedures for implementing and maintaining the equipment lineup necessary to perform the planned 3A CSP modification.
  3. Emphasize risk aspects of the proposed CT extension.
  4. Responding to unintended and unforeseen circumstances that may rely on CS system operability during the proposed Completion Time extension.
- Equipment walkdowns
- Progress Updates
- Increase operational and managerial scrutiny

In its letter dated February 16, 2018, the licensee revised its proposed markup of TS 3.2.1 to include the equipment that will be protected as compensatory measures. The licensee stated that the equipment would be guarded in accordance with plant procedure, OP-AA-102-1003, "Guarded Equipment."

The CS system primary function is to prevent long term containment pressure from exceeding design limits in the event of a LOCA or MSLB by spraying cool, borated water into the containment atmosphere. As such, failure of the CS system generally is not a major contributor to CDF and LERF. In its response to APLA RAI 01.b, the licensee provided additional PRA insights that further substantiate the very small increases in CDF and LERF as a result of one CS pump inoperable for the request to extend the CT. In its response to APLA RAI 01.b.1, the licensee indicated that the CS system is modeled in its PRA model only as an environmental qualification support system for the RHR suction valves (MOV-3-750 and MOV-3-751) and the containment sump level instrumentation. The licensee stated that because the CS system only affects these components, its impact on overall risk is minimal. To further address CS system impact on risk as modeled in the PRA, the licensee performed a quantitative analysis using a PRA model update and found the impact to be minimal. The licensee indicated that impact to risk does not result in any LERF increase and is only with respect to the baseline LERF, consequently the RG 1.177 risk metrics for delta CDF and delta LERF will be met.

Based on information from its review of RR No. 4, Amendment Nos. 263 and 258, and the licensee's letter dated February 16, 2018, the NRC staff concludes that the licensee's PRA model is acceptable to support the proposed change, because the licensee provided an adequate assessment of the change in risk for IE, IF, internal fires, and seismic events.

3.5.1.2 PRA Modeling

As discussed above, satisfaction of the fourth key principle of risk-informed decision-making may be demonstrated with reasonable assurance by comparing risk metrics that reflect the proposed TS changes to the numerical risk acceptance guidelines in RG 1.174 and RG 1.177.

RG 1.177, Section 2.3.3.1 states that to evaluate a TS change, specific systems or components involved in the change should be modeled in the PRA. The model should also be able to treat the alignments of components during periods when testing and maintenance are being carried out. Typically, LCOs and surveillance requirements relate to the system trains or components that are modeled in the system fault trees of a PRA. System fault trees should be sufficiently detailed to specifically include all the components for which surveillance tests and maintenance are performed and are to be evaluated.

For the preventive maintenance case, because such maintenance is planned to minimize plant risk consistent with the Maintenance Rule, 10 CFR 50.65(a)(4), the licensee used its PRA model with basic events set to zero for testing and maintenance and the 3A CS pump out of service (OOS). The NRC staff notes that, ideally, all risk metrics used in risk-informed license applications should be determined by adjusting an "average maintenance" PRA model (i.e., a PRA model that includes contributions from equipment maintenance unavailability). The use of a "zero maintenance" PRA model, which omits system maintenance unavailability contributions, in determining the RG 1.174 and RG 1.177 risk metrics introduces additional uncertainty into the analysis. However, the contribution from equipment maintenance unavailability to changes in risk depends on the likelihood of performing maintenance on other plant equipment in parallel with maintenance on the equipment whose CT is being extended. In Section 3.2.2 of letter dated December 18, 2017, the licensee described human performance actions where:

1. Prior to the start and during each shift of the proposed CT extension, a pre-job briefing will be conducted to reinforce expected human performance behaviors and bolster defense-in-depth barriers to human errors.
2. Operators and maintenance shift crews will be briefed on procedures for implementing and maintaining the equipment lineup necessary to perform the planned 3A CSP modification.
3. Operators will be additionally briefed on responding to unintended and unforeseen circumstances that may rely on CS system operability during the proposed CT extension.

The licensee further stated in Section 3.2.3 of its request that, "[d]uring the planned 3A CS Pump modification, no risk significant plant equipment will be removed from service and protective measures will be implemented to reduce the likelihood of challenges to risk significant equipment. As a result, the functional redundancy, independence and diversity currently described in the Turkey Point UFSAR will be maintained throughout the proposed Completion Time extension." Therefore, the likelihood of simultaneous maintenance actions is small. Maintenance that does occur will be evaluated and controlled by the licensee through its CRMP, and the human performance actions discussed above.

Because the magnitude of the additional uncertainty resulting from use of a "zero maintenance" PRA model is small, the NRC staff concludes that the licensee's use of a "zero maintenance" PRA model for the preventive maintenance case is acceptable.

The licensee's analysis evaluated the ICCDP and the ICLERP for the 3A CS pump being out of service for the requested 14-day CT. The NRC staff relied upon the results of the licensee's analysis for this configuration, as provided in the summary table below.

	ICCDP	ICLERP
Single 14 day TS 3.6.2.1 Entry	4.91E-11	2.30E-13

The above results include the impacts from IE and IF. As discussed in SE Section 3.5.1.1.3 above, for the fire model, CS pumps are assumed to fail and therefore, as a part of this risk evaluation, no impact was attributed. For seismic and other external hazards, the licensee assumed the contribution to be equal to the IE contribution, thereby doubling the ICCDP and ICLERP, which would result in the corresponding values below.

	ICCDP	ICLERP
Single 14 day TS 3.6.2.1 Entry	9.82E-11	4.60E-13

Section 2.4 of RG 1.177 states that a one-time TS CT change impact on plant risk is acceptable when the ICCDP is less than 1.0×10^{-6} and the ICLERP is less than 1.0×10^{-7}. Because the values are less than 1.0×10^{-6} and 1.0×10^{-7}, respectively, the ICCDP and ICLERP for 3A CS pump being inoperable for 14 days meet the acceptance guidelines for RG 1.177.

Section 2.5 of RG 1.177 states that in some cases, in support of a TS change, available alternatives are compared to justify the TS change. For changes in TS CTs, such cases primarily involve comparing the risk of shutting down with the risk of continuing power operation, given the plant is not meeting one or more TS LCOs. Such comparisons can be used to justify that the increase in at-power risk associated with the TS change is offset by the averting of some transition or shutdown risk. For the requested one-time TS change, the licensee stated that "the duration of the modifications were such that the 72 hour Completion Time allotted by TS 3.6.2.1, ACTION (a), would have been exceeded by several days had the modifications been performed at-power." The NRC staff concludes that the delta CDF and delta LERF values of 1.28E-09 and 6.00E-12, respectively, as discussed in Section 3.3.1.4 of the licensee's request dated December 18, 2017, for the increased representation of 3A CS pump being OOS for corrective maintenance, are very small with respect to the RG 1.174 acceptance metrics for risk and are acceptable.

Based on information from its review of RR No. 4, Amendment Nos. 263 and 258, and the licensee's letters dated December 18, 2017, and February 16, 2018, the NRC staff concludes that the licensee has adequately identified the impact of the one-time CT extension on plant risk and that impact to plant risk meets the acceptance guidelines for RG 1.177; therefore, the NRC staff finds that the licensee's Tier 1 evaluation is acceptable.

3.5.2 Tier 2: Avoidance of Risk-Significant Plant Configurations

RG 1.177 states that the licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out of

service consistent with the proposed TS change. The second tier evaluates the capability of the licensee to recognize and avoid risk-significant plant configurations that could result if equipment, in addition to that associated with the proposed change, is taken OOS simultaneously or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved.

In the its letter dated December 18, 2017, the licensee stated that "the deterministic defense in depth measures described above, along with online risk monitoring (below), are sufficient to ensure that configuration management will be maintained and that no risk significant plant configurations will occur during the proposed Completion Time extension." In APLA RAI 02, the NRC staff indicated that this reliance on the licensee's CRMP is more appropriate for the Tier 3 evaluation, which ensures that adequate programs and procedures are in place for identifying risk-significant plant configurations and taking appropriate actions to avoid such configurations.

Whereas the Tier 3 evaluation ensures the CRMP is adequate when maintenance is about to commence, the Tier 2 evaluation is meant to be an early evaluation (at the time of the licensee submittal requesting the action) to identify and preclude potentially high-risk plant configurations. One element described in Section 2.2.1 of RG 1.177 to assess whether the proposed TS change meets the defense-in-depth principle, includes consideration to preserve adequate capability of design features without over-reliance on programmatic activities as compensatory measures associated with the change in the licensing basis is avoided. To distinguish between Tier 2 and Tier 3 reliance on the CRMP, in APLA RAI 02, the NRC staff requested that the licensee identify for the proposed one-time CT extension any high-risk plant configurations that may occur and the compensatory measures that the licensee is implementing to ensure these configurations do not occur during the CT one-time extension. In response to APLA RAI 02, the licensee stated that cutsets from the risk calculation of the impact of having the CS pump out of service were reviewed to determine any configurations that should be avoided and to identify Tier 2 compensatory measures. The licensee confirmed that the top cutsets included failure events associated with the A-train of the CS system, CS system B-train, and emergency containment coolers. The NRC staff confirmed that all the identified failures have been included and addressed in adequate compensatory actions the licensee provided in Section 3.2.1 of the letter dated December 18, 2017.

Based on the licensee's response to APLA RAI 02, and review of the defense-in-depth measures described in SE Section 3.3, the NRC staff concludes that the licensee has demonstrated the ability to recognize and avoid risk-significant plant configurations, and therefore, the NRC staff finds that the licensee's Tier 2 evaluation is acceptable.

3.5.3 Tier 3: Risk-Informed Configuration Risk Management

The third tier assesses the licensee's program to ensure that the risk impact of OOS equipment is appropriately evaluated prior to performing any maintenance activity. The need for this third tier stems from the difficulty of identifying all possible risk-significant configurations under the second tier.

The licensee has a CRMP, which is described in Section 3.3.3 of its December 18, 2017, request. The CRMP is a proceduralized, risk-informed assessment process intended to manage the risk associated with planned and unplanned plant maintenance activities. The licensee stated that "online (i.e., at-power) risk monitoring encompasses an integrated review of known and anticipated plant conditions in order to uncover risk-significant plant configurations both during the work management process and for emergent plant operating conditions."

Section 2.3 of RG 1.177 states, in part, that "if the Tier 2 assessment demonstrates, with reasonable assurance, that there are no risk-significant configurations involving the subject equipment, the application of Tier 2 to the condition addressed by the proposed CT may not be necessary." In its response to APLA RAI 02, the licensee indicated that risk-significant equipment failures include the CS system A-train, CS system B-train, and the emergency containment coolers. The licensee included associated compensatory actions to address the potential risk-significant equipment. Given the very low orders-of-magnitude for ICCDP and ICLERP that were quantified, the NRC staff finds that the licensee's CRMP for addressing Tier 3, as described in RG 1.177, in addition to the human performance measures described in Section 3.2.2 of its December 18, 2017, request, is acceptable for assessing and managing OOS equipment to uncover risk-significant plant equipment outage configurations in a timely manner during normal plant operations.

3.6 Key Principle 5: Monitor the Impact of the Proposed Change

RG 1.174 and RG 1.177 establishes the need for an implementation and monitoring program to ensure that extensions to TS CTs do not degrade operational safety over time and that no adverse degradation occurs due to unanticipated degradation or common cause mechanisms. An implementation and monitoring program is intended to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of structures, systems, and components impacted by the change. RG 1.174 states that monitoring performed under the Maintenance Rule, 10 CFR 50.65, can be used when the monitoring performed is sufficient for the structures, systems, and components affected by the risk-informed application.

In Section 3.1 of its request dated December 18, 2017, the licensee stated that no such change to the Maintenance Rule Monitoring Plan is proposed. Furthermore, the licensee stated that the 3A CS pump will continue to be maintained and monitored such that exceeding the applicable Maintenance Rule performance criteria will result in increased monitoring and goal setting, in accordance with the requirements at 10 CFR 50.65(a)(1).

The NRC staff finds that the implementation and monitoring program for the proposed TS change described by the licensee is consistent with Key Principle 5 of RG 1.177 and are acceptable.

3.7 Technical Conclusion

As discussed in Section 3.1 of this SE, an acceptable approach for making risk-informed decisions about proposed TS changes, including both permanent and temporary changes, is to show that the proposed changes meet the five key principles stated in RG 1.174, Section 2, and RG 1.177, Section B. The NRC staff determined in Sections 3.2 through 3.6 of this SE that the licensee's proposed one-time extension of the CT for the CS system from 72 hours to 14 days meets the five key principles. As a result, the NRC staff finds that the proposed changes to TS LCO 3.6.2.1 are consistent with the requirements of 10 CFR 50.36(c)(2) and are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the State of Florida official (Ms. Cynthia Becker, M.P.H., Chief of the Bureau of Radiation Control, Florida Department of Health) on March 9, 2018 (ADAMS Accession No. ML18068A499), of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding, which was published in the FR on January 30, 2018 (83 FR 4285), that the amendment involves NSHC, and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributors: Adrienne Driver
Gerard Purciarello

Date: April 3, 2018

SUBJECT: TURKEY POINT NUCLEAR GENERATING UNIT NO. 3 – ISSUANCE OF AMENDMENT REGARDING THE ONE-TIME EXTENSION OF THE CONTAINMENT SPRAY SYSTEM COMPLETION TIME (EPID L-2017-LLA-0423) DATED APRIL 3, 2018

DISTRIBUTION:
PUBLIC
LPL2-2 R/F
RidsACRS_MailCTR Resource
RidsRgn2MailCenter Resource
RidsNrrPMTurkeyPoint Resource
RidsNrrDorlLpl2-2 Resource
RidsNrrLABClayton Resource
RidsNrrDraApla Resource
RidsNrrDssScpb Resourc
RidsNrrDssSrxb Resource
RidsNrrDssStsb Resource
RBeaton, NRR
MChernoff, NRR
ADriver, NRR
GPurciarello, NRR

ADAMS Accession No.: ML18075A348 *by memo **by email

OFFICE	NRR/DORL/LPL2-2/PM	NRR/DORL/LPL2-2/LA	NRR/DRA/APLA/BC*
NAME	MWentzel	BClayton	SRosenberg
DATE	03/20/18	03/20/18	03/15/18
OFFICE	NRR/DSS/SCPB/BC*	NRR/DSS/SRXB/BC(A)	NRR/DSS/STSB/BC
NAME	RDennig	JWhitman	VCusumano
DATE	03/06/18	03/20/18	03/23/18
OFFICE	OGC (NLO)	NRR/DORL/LPL2-2/BC(A)	NRR/DORL/LPL2-2/PM
NAME	MYoung	BTindell	MWentzel
DATE	03/29/18	04/03/18	04/03/18

OFFICIAL RECORD COPY