FPL

June 13, 2019
L-2019-053
10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington D C 20555-0001

RE: Turkey Point Nuclear Plant, Unit 3 and 4
Docket Nos. 50-250 and 50-251
Renewed Facility Operating Licenses DPR-31 and DPR-41

License Amendment Request 269, Modify Reactor Trip System (RTS) Turbine Trip Instrumentation Requirements and Resolve Two Non-Conservative Requirements

Pursuant to 10 CFR Part 50.90, Florida Power & Light Company (FPL) hereby requests amendments to Renewed Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Nuclear Plant Units 3 and 4 (Turkey Point), respectively. The proposed license amendments revise the Turkey Point Technical Specifications (TS) by modifying the Reactor Trip System (RTS) turbine trip instrumentation requirements to align with the permissive interlock, P-7. The proposed license amendments additionally resolve two non-conservative requirements, consistent with NRC Administrative Letter 98-10, Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety, (NUDOCS Accession No. 9812280273).

The enclosure to this letter provides FPL's evaluation of the proposed changes. Attachment 1 to the enclosure provides a mark-up of the existing TS pages to show the proposed changes. No changes are proposed to the Turkey Point TS Bases.

FPL has determined that the proposed changes do not involve a significant hazards consideration pursuant to 10 CFR 50.92(c), and there are no significant environmental impacts associated with the proposed changes. The Turkey Point Onsite Review Group has reviewed the proposed license amendments. In accordance with 10 CFR 50.91(b)(1), a copy of the proposed license amendments are being forwarded to the State designee for the State of Florida.

FPL requests that the proposed changes are processed as a normal license amendment request with approval within one year of the submittal date. Once approved, the amendments will be implemented within 90 days.

This letter contains no regulatory commitments.

Should you have any questions regarding this submission, please contact Mr. Robert Hess, Turkey Point Licensing Manager, at 305-246-4112.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on the 17th day of June 2019.

Sincerely,

B. Stamp

Brian Stamp
Site Director, Turkey Point Nuclear Plant

Enclosure
Attachments

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, Turkey Point Nuclear Plant
USNRC Senior Resident Inspector, Turkey Point Nuclear Plant
Ms. Cindy Becker, Florida Department of Health

EVALUATION OF THE PROPOSED CHANGES

Turkey Point Nuclear Plant Unit 3 and Unit 4
License Amendment Request 269, Modify Reactor Trip System (RTS) Turbine Trip Instrumentation Requirements and Resolve Two Non-Conservative Requirements

1.0 SUMMARY DESCRIPTION

Florida Power & Light Company (FPL) requests amendments to Renewed Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Nuclear Plant Units 3 and 4 (Turkey Point), respectively. The proposed license amendments revise the Turkey Point Technical Specifications (TS) by modifying the Reactor Trip System (RTS) turbine trip instrumentation requirements to align with the permissive interlock, P-7. The proposed license amendments additionally resolve two non-conservative TS requirements, consistent with NRC Administrative Letter (AL) 98-10, Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety (Reference 6.1).

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

2.1.1 Reactor Trip System (RTS)

The RTS is designed to trip the reactor in order to prevent core damage caused by departure from nucleate boiling (DNB) and to preserve Reactor Coolant System (RCS) integrity during limiting fault conditions. The RTS monitors neutron flux, RCS pressure, temperature and flow, pressurizer water level, steam generator water level, feedwater flow, and turbine generator operational status. The RTS provides the primary trip functions for the over-power ΔT, over-temperature ΔT and nuclear overpower reactor trips. The operating region below these trip settings is designed such that no combination of power, temperature and pressure can result in DNBR less than the safety analysis limit value with all reactor coolant pumps in operation. The RTS provides additional reactor trip functions such as the high and the low pressurizer pressure trips, safety injection trip, turbine trip, etc., which serve as back up to the primary trip functions for specific accident conditions and mechanical failures. The RTS ensures that plant operating conditions never approach safety analysis limits.

Certain power range reactor trip channels are automatically bypassed at low power levels where they are not required to assure safety. Nuclear source range and intermediate range trips are specifically provided for protection at low power or subcritical operation and are manually bypassed at higher power levels.

2.1.2 Permissive Interlock, P-7

The RTS (and ESF actuation) logic circuitry employs permissive interlocks which provide input to protection system functions that are necessary to assure safety. Amongst the permissive interlocks, the RTS utilizes permissive P-7, which is set at approximately 10% Rated Thermal Power (RTP). The following reactor trips are enabled above the P-7 permissive:

- Pressurizer low-pressure reactor trip
- Pressurizer high-level reactor trip
- 4KV Bus A & B under-voltage reactor trip
- RCP under-frequency reactor trip
- RCP open breaker (2 out of 3 pumps) reactor trip
- Reactor coolant loop low-flow (2 out of 3 loops) reactor trip
- Turbine trip

Upon decreasing reactor power, the above listed trips are automatically blocked below the P-7 permissive.

2.1.3 Reactor Trip on Turbine Trip Protective Feature

The 'reactor trip on turbine trip' protective feature anticipates the loss of heat removal capability of the secondary system following a turbine trip in order to minimize the pressure and temperature transient on the reactor. The reactor trip on turbine trip is actuated when two out of three emergency trip header oil pressure sensors indicate pressure below a preset value established by safety analyses or two out of two turbine stop valve position sensors indicate both valves closed. Following a reactor trip on turbine trip, sensible heat stored in the reactor coolant is removed without actuating the steam generator Code safety valves by means of controlled steam bypass to the condenser and by feedwater injection to the steam generators in order to reach RCS no-load conditions.

Though the Turkey Point units are capable of withstanding a 50% loss of load without tripping the reactor, the reactor trip on turbine trip is enabled at the P-7 permissive (~10% RTP). The reactor trip on turbine trip is automatically blocked below P-7 where intermediate range neutron flux permissives enable low power protective features and where the reactor control system and the steam dump system will automatically control the reactor to zero power conditions.

The reactor trip on turbine trip function is not credited in Turkey Point accident analyses but is considered an anticipatory trip which serves to enhance overall reactor protection system reliability. In the event of a turbine trip from full power without an immediate reactor trip, protection circuits associated with coolant conditions directly tied to core limits will actuate to prevent core damage. However, the reactor trip on turbine trip circuitry satisfy the applicable protection system standards for separation, redundancy, testability and thereby reliability.

2.1.4 Engineered Safety Features Actuation System (ESFAS)

The ESFAS system senses process variables in the reactor coolant, steam, reactor containment and auxiliary systems, and actuates the Engineered Safety Features which ensure that predetermined limits will not be exceeded. The ESFAS actuation channels combine redundant sensors, channel separation and coincident trip logic to ensure system reliability such that a single failure cannot defeat the channel function. The ESFAS will actuate the safety injection, containment isolation, emergency containment cooling and/or the containment spray system depending upon the type and severity of the plant condition.

2.2 Current Requirements / Description of the Proposed Change

2.2.1 TS 3.3.1, Table 3.3-1, Reactor Trip System Instrumentation, specifies the applicable MODES for Functional Unit (FU) 15.a, Turbine Trip (Above P-7) Emergency Trip Header Pressure, and FU 15.b, Turbine Trip (Above P-7) Turbine Stop Valve Closure, instrument channels.

The proposed change modifies the applicable MODES of TS 3.3.1, Table 3.3-1, to require operability of the FU 15.a and FU 15.b instrument channels only in MODE 1 when above the P-7 permissive. The proposed change is as follows:

TABLE 3.3-1
REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	...	...	...	APPLICABLE MODES	...

15. Turbine Trip (Above P-7)					
a. Emergency Trip Header Pressure	…	...	…	1 ***(Above P-7)***	…
b. Turbine Stop Valve Closure	…	..	...	1 ***(Above P-7)***	…

2.2.2 TS 3.3.1, Table 4.3-1, Reactor Trip System Instrumentation Surveillance Requirements, specifies the Trip Actuating Device Operational Test (TADOT) requirements for the FU 15.a and FU 15.b instrument channels. NOTE (1) and NOTE (10) of TS 3.3.1, Table 4.3-1, provide additional information relating to the TADOT requirement for the FU 15.a and FU 15.b channels.

The proposed change modifies the surveillance requirements of TS 3.3.1, Table 4.3-1, to require TADOT performance of the FU 15.a and FU 15.b channels prior to reaching the P-7 permissive, and modifies the surveillance required MODES with a new table notation denoted by four asterisks (****) to specify MODE 1 above the P-7 permissive. The proposed change additionally deletes Note 1 from the FU 15.a and FU 15.b channels and adds to Note 10 a statement requiring TADOT of the FU 15.a and FU 15.b channels whenever the Unit has been in MODE 3 if not performed within the previous 31 days. The proposed change is as follows:

TABLE 4.3-1
REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQ'S

FUNCTIONAL UNIT	…	…	…	TRIP ACTUATING DEVICE OPERATIONAL TEST	…	MODES FOR WHICH SURVEILLANCE IS REQUIRED
15. Turbine Trip						
a. Emergency Trip Header Pressure	…	…	…	***Prior to P-7***~~S/U~~(~~1,~~10)	…	1**********
b. Turbine Stop Valve Closure	…	…	…	***Prior to P-7***~~S/U~~(~~1,~~10)	…	1**********

TABLE 4.3-1
TABLE NOTATIONS

******* Above P-7 (Low Power Reactor Trips Block Interlock) Setpoint.***

Note (10) ***Required whenever the Unit has been in MODE 3 if not performed within previous 31 days.*** Setpoint verification is not applicable.

2.2.3 ACTION 13 of TS 3.3.1, Table 3.3-1, Reactor Trip System Instrumentation, specifies requirements for an inoperable FU-5, Over-temperature-ΔT, FU-6, Over-power ΔT, or FU-9, Pressurizer Level-High, instrument channel.

The proposed change modifies ACTION 13 of TS 3.3.1, Table 3.3-1, by removing the authorization to place an inoperable FU-5, FU-6, or FU-9 channel in bypass status to allow performance of a subsequent required digital channel operational test (DCOT). The proposed change is as follows:

ACTION 13 With the number of OPERABLE channels one less than the Total number of channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours. ~~*For subsequent required DIGITAL*~~

~~***CHANNEL OPERATIONAL TESTS the inoperable channel may be placed in bypass status for up to 4 hours.***~~

2.2.4 ACTION 25 of TS 3.3.2, Table 3.3-2, Engineered Safety Features Actuation System Instrumentation, specifies the requirements for an inoperable FU-1.f, *Safety Injection - Steam Line Flow-High Coincident with* T_{avg}*-Low*, or FU-4.d, *Steam Line Isolation - Steam Line Flow-High Coincident with* T_{avg}*-Low,* ESFAS instrument channel.

The proposed change modifies ACTION 25 of TS 3.3.2, Table 3.3-2, by removing the authorization to place an inoperable FU-1.f or FU-4.d channel in bypass status to allow performance of a subsequent required DCOT. The proposed change is as follows:

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

2.3 Reason for the Proposed Change

The proposed license amendments serve to align the requirements for the RTS turbine trip instrumentation with the plant conditions that are necessary for the protective feature to function (i.e. above P-7). The proposed license amendments additionally resolve two non-conservative TS requirements, consistent with NRC AL 98-10 (Reference 6.1).

3.0 TECHNICAL EVALUATION

3.1 Modify FU 15.a, Turbine Stop Valve, and FU 15.b, Emergency Trip Header, Requirements

3.1.1 TS 3.3.1, Table 3.3-1, Applicable MODES

The proposed change modifies the requirements for RTS turbine trip channels FU 15.a, Turbine Trip (Above P-7) Emergency Trip Header Pressure, and FU 15.b, Turbine Trip (Above P-7) Turbine Stop Valve Closure, such that operability is not required below the P-7 permissive (~10% RTP). When above P-7, the FU 15.a and FU 15.b instrument channels provide the ‘reactor trip on turbine trip’ protective feature whenever 2 of the 3 emergency trip header oil pressure signals indicate pressure below a preset value established by safety analyses or when 2 of the 2 turbine stop valve position signals indicate both valves closed. TS 3.3.1, Table 3.3-1, currently requires the FU 15.a and FU 15.b channels to be operable in MODE 1. As discussed in Section 2.1.3 however, the reactor trip on turbine trip protective feature is not enabled until reactor power reaches the P-7 permissive. Subjecting the FU 15.a and FU 15.b channels to operability requirements during plant conditions when they are not required to function is unreasonable since the practice provides no benefit to safety and can distract plant resources from more safety significant activities. The proposed change to require FU 15.a and FU 15.b channel operability in MODE 1 when above the P-7 permissive is appropriate since the change aligns the operability requirements with the plant conditions that are necessary for the protective feature to function. The proposed change is consistent with Table 3.3.1-1 of Standard Technical Specifications (STS) 3.3.1, Reactor Trip System Instrumentation (Reference 6.2), which requires operability of the RTS turbine trip channels, FU 16.a and FU 16.b, at the power level presumed for RTS turbine trip functionality [P-9], and is thereby reasonable.

3.1.2 TS 3.3.1, Table 4.3-1, TADOT and Surveillance MODES Requirements

The proposed change modifies the surveillance requirements of TS 3.3.1, Table 4.3-1, such that TADOT of the FU 15.a and FU 15.b channels will be performed prior to reaching the P-7 permissive. TS 3.3.1, Table 4.3-1, specifies “S/U” (i.e. Startup) in the TADOT column for the FU 15.a and FU 15.b channels. Compliance with TS 3.3.1, Table 4.3-1, requires performance of the TADOT prior to entering MODE 2. However, TADOT of the FU 15.a and FU 15.b channels can present operational challenges when approaching startup since the only heat input to the RCS are primarily decay heat and the RCPs, thereby leaving RCS pressure vulnerable to the secondary steam demand that could occur during TADOT performance. Moreover, subjecting the FU 15.a and FU 15.b channels to surveillance requirements well before the protective feature is required to function is unreasonable since the practice provides no benefit to safety and can distract plant resources from more safety significant activities. Modifying TS 3.3.1, Table 4.3-1, such that TADOT of the FU 15.a and FU 15.b channels is required prior to reaching the P-7 permissive aligns the surveillance requirement with the plant conditions that necessitate the protective feature. In addition, the proposed change would allow TADOT performance when sufficient nuclear heat is available to assure safety while maintaining the requirement to demonstrate functional capability before the protective feature is required to be operable. To align the surveillance required MODES with the proposed TADOT requirement, the proposed change replaces ”S/U” (i.e. Startup) in the TADOT column of TS 3.3.1, Table 4.3-1 for the FU 15.a and FU 15.b channels to become “Prior to P-7”. The proposed change adds four asterisks (****) in the MODES FOR WHICH SUREVILLANCE IS REQUIRED column for the FU 15.a and FU 15.b channels and adds a new note denoted by the four asterisks (****) to the TABLE NOTATIONS clarifying that the surveillance requirements for the FU 15.a and FU 15.b channels are applicable when above the P-7 setpoint. The proposed change also adds to existing “Note 10” a statement that TADOT is required whenever the Unit has been in MODE 3 if not performed within the previous 31 days. The proposed change to Note 10 allows deletion of Note 1 in the TADOT column of TS 3.3.1, Table 4.3-1 for the FU 15.a and FU 15.b channels since the content of Note 1 is being repeated in the revised Note 10. The proposed change to Note 10 requiring TADOT performance whenever the Unit has been in MODE 3 precludes testing prompted by planned power reductions in MODE 1 and 2. The proposed change is consistent with SR 3.3.1.15 of STS 3.3.1, Reactor Trip System Instrumentation (Reference 6.2), which imposes TADOT on the FU 16.a and FU 16.b channels prior to the power level presumed for RTS turbine trip functionality [P-9] when the Unit has been in MODE 3 if not performed within the previous 31 days, and is thereby reasonable.

3.2 Resolve Potentially Non-Conservative Requirements

3.2.1 TS 3.3.1, Table 3.3-1, ACTION 13

The proposed change removes from ACTION 13 of TS 3.3.1, Table 3.3-1, the authorization to place an inoperable FU-5, Over-temperature-ΔT, FU-6, Over-power ΔT, or FU-9, Pressurizer Level-High, channel in bypass in order to perform DCOT testing of an operable channel. ACTION 13 specifies the requirements for an inoperable FU-5, FU-6 or FU-9 instrument channel. Table 3.3-1 of TS 3.3.1 specifies that the FU-5, FU-6 and FU-9 functions each have a total of three channels, a minimum channels operable requirement of two, and a 2 out of 3 channels-to-trip logic circuitry. ACTION 13 allows continued plant operation with one inoperable channel provided the inoperable channel is placed in the tripped condition within 6 hours. ACTION 13 also allows an inoperable channel to be placed in bypass for up to 4 hours in order to perform DCOT testing on an operable FU-5, FU-6 or FU-9 channel. However, since FU-5, FU-6 and FU-9 each employ a 2 out of 3 channels-to-trip logic to perform their respective trip function, placing an inoperable channel in bypass while DCOT testing an associated operable channel could result in a loss of function. The proposed change removes from ACTION 13, authorization to place an inoperable FU-5, FU-6 or FU-9 channel in bypass in order to perform DCOT on an associated FU-5, FU-6 or FU-9 channel. Consistent with NRC Administrative Letter 98-10 (Reference 6.1), FPL established interim corrective measures which disallow placing an inoperable FU-5, FU-6 or FU-9 channel in bypass. The interim measures will remain in effect pending implementation of the proposed license amendments.

3.2.2 Table 3.3-2, TS 3.3.2, ACTION 25

The proposed change removes from ACTION 25 of TS 3.3.2, Table 3.3-2, the authorization to place an inoperable FU-1.f, Safety Injection - Steam Line Flow-High Coincident with T_{avg}-Low, or FU-4.d, Steam Line Isolation - Steam Line Flow-High Coincident with T_{avg}-Low, ESFAS instrument channel in bypass status in order to perform DCOT on an operable FU-1.f or FU-4.d instrument channel. ACTION 25 specifies requirements for an inoperable FU-1.f or FU-4.d instrument channel. Table 3.3-2 of TS 3.3.2 specifies that the FU-1.f and FU-4.d functions each have one channel per RCS loop and a 2 out of 3 (RCS loops) channels-to-trip logic circuitry. ACTION 25 allows STARTUP and/or POWER OPERATION to proceed with one less than the total number of RCS loop channels provided the inoperable channel is placed in the tripped condition within 6 hours. ACTION 25 also allows an inoperable FU-1.f or FU-4.d channel to be placed in bypass up to 4 hours during DCOT testing of a FU-1.f or FU-4.d channel associated with another RCS loop. However, since FU-1.f or FU-4.d each employ a 2 out of 3 channels-to-trip logic to perform their respective ESFAS actuation function, placing an inoperable channel in bypass while DCOT testing an associated operable channel could result in a loss of function. The proposed change removes from ACTION 25, authorization to place an inoperable FU-1.f or FU-4.d channel in bypass in order to perform DCOT on an associated operable FU-1.f or FU-4.d channel. Consistent with NRC Administrative Letter 98-10 (Reference 6.1), FPL established interim corrective measures which disallow placing an inoperable FU-1.f or FU-4.d channel in bypass. The interim measures will remain in effect pending implementation of the proposed license amendments.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

- 10 CFR 50.36(c)(3) states that 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.

- 1967 Proposed General Design Criteria (GDC) 6 states that the reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated.

- 1967 Proposed GDC 12 states that instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables

- 1967 Proposed GDC 14 states that core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

- 1967 Proposed GDC 15 states that protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

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

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

- 1967 Proposed GDC 26 states that the reactor protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electrical power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

- 1967 Proposed GDC 31 states that the reactor protection system shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

- 1967 Proposed GDC 37 states that engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. Such engineered safety features shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary, assuming unobstructed discharge from both ends.

- 1967 Proposed GDC 38 states that all engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public.

- 1967 Proposed GDC 41 states that engineered safety features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public.

- GDC 12 of 10 CFR 50, Appendix A, states that instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables.

- Administrative Letter (AL) 98-10 states that imposing administrative controls in response to improper or inadequate TS is considered an acceptable short-term corrective action, and that following the imposition of administrative controls, an amendment to the TS, with appropriate justification and schedule, will be submitted in a timely fashion.

The proposed license amendments comply with the 10 CFR 50.36(c)(3) requirements and do not alter the manner in which Turkey Point will be operated and maintained consistent with 1967 Proposed GDC(s) 6, 12, 14, 15, 19, 20, 26, 31, 37, 38, 41 and GDC 12. All applicable regulatory requirements will continue to be satisfied as a result of the proposed license amendments.

4.2 No Significant Hazards Consideration

The proposed license amendments revise the Turkey Point Technical Specifications (TS) by modifying the Reactor Trip System (RTS) turbine stop valve closure and turbine emergency trip header pressure instrumentation requirements to align with the permissive interlock, P-7 (~10% RTP). The proposed license amendments additionally resolve two potentially non-conservative requirements, consistent with NRC Administrative Letter 98-10 (Reference 6.1). As required by 10 CFR 50.91(a), FPL has evaluated the proposed change using the criteria in 10 CFR 50.92 and has determined that the proposed change does not involve a significant hazards consideration. An analysis of the issue of no significant hazards consideration is presented below:

(1) Do the proposed amendments involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed amendments modify the mode of applicability and surveillance requirements for the Reactor Trip System (RTS) turbine trip instrumentation such that operability is required in MODE 1 when above the permissive interlock, P-7, and satisfactory surveillance testing is required prior to reaching MODE 1 above P-7 whenever the Unit has been in MODE 3. Aligning the operability requirements with the plant conditions required for the protective feature to function neither changes the manner in which operability will be determined nor the manner in which the equipment will be operated and maintained. No change to the RTS turbine trip instrumentation is proposed and the equipment will remain capable of performing as required upon implementation of the proposed amendments. No changes are proposed to any safety analysis inputs or assumptions. The proposed change additionally resolves two non-conservative TS requirements consistent with NRC Administrative Letter 98-10, and thereby cannot adversely affect the likelihood or the outcome of any design basis accident.

Therefore, this proposed change does not represent a significant increase in the probability or consequences of an accident previously evaluated.

(2) Do the proposed amendments create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed amendments align the RTS turbine trip channel requirements with the plant conditions required for the protective feature to function (i.e. P-7). The proposed change establishes RTS turbine trip channel operability in MODE 1 when above P-7 and requires surveillance testing prior to MODE 1 above P-7 whenever the Unit has been in MODE 3. The inputs and assumptions to safety analyses remain unchanged as a result of the proposed change since no physical change to plant equipment is proposed and the requirement to demonstrate operability prior to the plant conditions necessitating the protective feature remains unchanged. As such, the proposed change cannot introduce new equipment failure modes, cannot change the types or amount of effluent that may be released off-site, and cannot increase individual or cumulative occupational exposures that would result from any accident. The proposed change additionally resolves two non-conservative requirements consistent with NRC Administrative Letter 98-10, and thereby cannot create a new or different kind of accident.

Therefore, the proposed amendments do not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Do the proposed amendments involve a significant reduction in a margin of safety?

Response: No

The proposed amendments align the RTS turbine trip channel requirements with the plant conditions required for the protective feature to function (i.e. P-7) by establishing RTS turbine trip channel operability in MODE 1 when above P-7 and requiring surveillance testing prior to MODE 1 above P-7 whenever the Unit has been in MODE 3. The proposed amendments additionally resolve two non-conservative TS requirements consistent with NRC Administrative Letter 98-10. The proposed changes do not affect any plant operating margins or the reliability of equipment credited in safety analyses and no changes are proposed to any safety analysis assumptions, safety limits, or limiting safety system settings.

Therefore, the proposed amendments do not involve a significant reduction in a margin of safety.

Based upon the above analysis, FPL concludes that the proposed license amendments do not involve a significant hazards consideration, under the standards set forth in 10 CFR 50.92, “Issuance of Amendment,” and accordingly, a finding of “no significant hazards consideration” is justified.

4.3 <u>Conclusion</u>

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) 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.

5.0 ENVIRONMENTAL CONSIDERATION

FPL has evaluated the proposed amendment for environmental considerations. The review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set for in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

6.0 REFERENCES

6.1 NRC Administrative Letter 98-10, Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety, December 29, 1998 (NUDOCS Accession No. 9812280273).

6.2 NUREG-1431, Standard Technical Specifications - Westinghouse Plants, Revision 4.0, Volume 1, Specifications (Accession No. ML12100A222)

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION PAGE (MARKUP)

(9 pages follow)

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION PAGE (MARKUP)

This page is for Information only.
There are no changes to this page.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirement specified in Table 4.3-1.

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION PAGE (MARKUP)

This page is for Information only.
There are no changes to this page.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1##, 2	2
3. Intermediate Range, Neutron Flux	2	1	2	1##, 2	3
4. Source Range, Neutron Flux					
a. Startup	2	1	2	2#	4
b. Shutdown**	2	0	2	3, 4, 5	5
c. Shutdown	2	1	2	3*, 4*, 5*	9
5. Overtemperature ΔT	3	2	2	1, 2	13
6. Overpower ΔT	3	2	2	1, 2	13
7. Pressurizer Pressure-Low (Above P-7)	3	2	2	1	6
8. Pressurizer Pressure--High	3	2	2	1, 2	6
9. Pressurizer Water Level--High (Above P-7)	3	2	2	1	13
10. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop	2/loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop	2/loop	1	6

TURKEY POINT – UNITS 3 & 4

3/4 3-2

AMENDMENT NOS. 140 AND 135

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PROPOSED TECHNICAL SPECIFICATION PAGE (MARKUP)

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Steam Generator Water Level--Low-Low	3/stm. gen.	2/stm. gen.	2/stm. gen.	1, 2	6
12. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	2 stm. gen. level and 2 stm./feed-water flow mismatch in each stm. gen.	1 stm. gen. level coin-cident with 1 stm./feed-water flow mismatch in same stm. gen.	1 stm. gen. level and 2 stm./feed-water flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen.	1, 2	6
13. Undervoltage--4.16 KV Busses A and B (Above P-7)	2/bus	1/bus on both busses	2/bus	1	12
14. Underfrequency-Trip of Reactor Coolant Pump Breaker(s) Open (Above P-7)	2/bus	1 to trip RCPs***	2/bus	1	11
15. Turbine Trip (Above P-7)					Add
a. Emergency Trip Header Pressure	3	2	2	1 (Above P-7)	12
b. Turbine Stop Valve Closure	2	2	2	1 (Above P-7)	12

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PROPOSED TECHNICAL SPECIFICATION PAGE (MARKUP)

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 11 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ACTUATION LOGIC TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 13 - With the number of OPERABLE channels one less than the Total number of channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours. ~~For subsequent required DIGITAL CHANNEL OPERATIONAL TESTS the inoperable channel may be placed in bypass status for up to 4 hours.~~

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PROPOSED TECHNICAL SPECIFICATION PAGE (MARKUP)

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
12. Steam Generator Water Level--Low Coincident with Steam/Feedwater Flow Mismatch	SFCP	SFCP(a), (b)	SFCP(a), (b)	N.A.	N.A.	1, 2
13. Undervoltage – 4.16 kV Busses A and B	N.A.	SFCP	N.A.	N.A.	N.A.	1
14. Underfrequency – Trip of Reactor Coolant Pump Breakers(s) Open	N.A.	SFCP	N.A.	N.A.	N.A.	1
15. Turbine Trip				Prior to P-7		Add 4 asterisks "****"
a. Emergency Trip Header Pressure	N.A.	SFCP(a), (b)	N.A.	~~S/U~~(~~1,~~ 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	SFCP	N.A.	~~S/U~~(~~1,~~ 10)	N.A.	1
16. Safety Injection Input from ESF	N.A.	N.A.	N.A.	SFCP	N.A.	1, 2
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	SFCP(4)	SFCP	N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7 (includes P-10 input and Turbine Inlet Pressure)	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	SFCP(4)	SFCP	N.A.	N.A.	1

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PROPOSED TECHNICAL SPECIFICATION PAGE (MARKUP)

**** Above P-7 (Low Power Reactor Trips Block Interlock) Setpoint.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

** Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

*** Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(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 (NTS) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTS are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field settings) to confirm channel performance. The NTS and methodologies used to determine the as-found and the as-left tolerances are specified in UFSAR Section 7.2.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power level indication above 15% of RATED THERMAL POWER (RTP). Adjust excore channel gains consistent with calorimetric power level if the absolute difference is greater than 2%. Below 70% RTP, downward adjustments of NIS excore channel gains to match a lower calorimetric power level are not required. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) This table Notation number is not used.

(6) Incore-Excore Calibration, above 75% of RATED THERMAL POWER (RTP). If the quarterly surveillance requirement coincides with sustained operation between 30% and 75% of RTP, calibration shall be performed at this lower power level. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested in accordance with the Surveillance Frequency Control Program.

(8) DELETED

(9) Quarterly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissive P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Quarterly surveillance shall include verification of the High Flux at Shutdown Alarm Setpoint of 1/2 decade above the existing count rate.

Add "Required whenever Unit has been in MODE 3 if not performed within previous 31 days."

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the OPERABILITY of the undervoltage and shunt trip attachment of the Reactor Trip Breakers.

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PROPOSED TECHNICAL SPECIFICATION PAGE (MARKUP)

This page is for information only.
There are no changes to this page.

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Line flow--High Coincident with:	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3*	15
Steam Generator Pressure--Low	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2. 3*	15
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3*	25
2. Containment Spray					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Containment Pressure--High-High Coincident with:	3	2	2	1, 2, 3	15
Containment Pressure--High	3	2	2	1, 2, 3	15
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

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PROPOSED TECHNICAL SPECIFICATION PAGE (MARKUP)

This page is for information only.
There are no changes to this page.

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
d. Steam Line Flow--High Coincident with: Steam Generator Pressure--Low	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3	15
	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2, 3	15
or T_{avg}--Low	1/Loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3	25
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
b. Safety-Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Steam Generator Water Level -- High-High# # # #	3/steam generator	2/steam generator in any operating steam generator	2/steam generator in any operating steam generator	1, 2, 3	15
6. Auxiliary Feedwater# # #					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20

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PROPOSED TECHNICAL SPECIFICATION PAGE (MARKUP)

TABLE 3.3-2 (Continued)

TABLE NOTATION (Continued)

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

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

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

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

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

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