ENCLOSURE 2

TURKEY POINT NUCLEAR PLANT, UNITS 3 AND 4
IMPROVED TECHNICAL SPECIFICATIONS (ITS) SUBMITTAL
VOLUMES 1 THROUGH 16

(3267 TOTAL PAGES, INCLUDING COVER SHEETS)

ENCLOSURE 2

VOLUME 1

TURKEY POINT NUCLEAR GENERATING STATION UNIT 3 AND UNIT 4

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

APPLICATION OF SELECTION CRITERIA TO THE TECHNICAL SPECIFICATIONS

REVISION 0

**APPLICATION OF SELECTION CRITERIA TO THE
TURKEY POINT NUCLEAR GENERATING STATION UNIT 3 AND UNIT 4
TECHNICAL SPECIFICATIONS**

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APPLICATION OF SELECTION CRITERIA TO THE TURKEY POINT NUCLEAR GENERATING STATION UNIT 3 AND UNIT 4 TECHNICAL SPECIFICATIONS

1. INTRODUCTION

The purpose of this document is to confirm the results of the Westinghouse Owners Group application of the Technical Specification selection criteria on a plant specific basis for the Turkey Point Nuclear Generating Station Unit 3 and Unit 4 (hereinafter PTN). Florida Power & Light (hereinafter FPL) has reviewed the application and confirmed the applicability of the selection criteria to each of the Technical Specifications utilized in report WCAP-11618, "Methodically Engineered Restructured and Improved Technical Specifications, MERITS Program - Phase II Task 5, Criteria Application" (Reference 1) including Addendum 1, the NRC Staff Review of NSSS Vendor Owners Groups Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications, Wilgus/Murley letter dated May 9, 1988, and as revised in NUREG-1431, Revision 5.0, "Standard Technical Specifications, Westinghouse Plants" (Reference 2), and applied the criteria to each of the current PTN Technical Specifications. Additionally, in accordance with the NRC Final Policy Statement (Reference 3), this confirmation of the application of selection criteria includes confirming the risk insights from Probabilistic Risk Assessment (PRA) evaluations, provided in Reference 1, as applicable to PTN.

2. SELECTION CRITERIA

FPL has utilized the selection criteria provided in the NRC Final Policy Statement on Technical Specification Improvements of July 22, 1993 (Reference 3), to develop the results contained in the attached matrix. PRA insights as used in the Westinghouse Owners Group submittal were utilized, confirmed by FPL, and are discussed in the next section of this report. The selection criteria and discussion provided in Reference 3 are as follows: (I assume all of the below are quotes so I didn't mess with the language, which is lacking in some cases…)

<u>Criterion 1</u> Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

<u>Discussion of Criterion 1</u>

A basic concept in the adequate protection of the public health and safety is the prevention of accidents. Instrumentation is installed to detect significant abnormal degradation of the reactor coolant pressure boundary so as to allow operator actions to either correct the condition or to shut down the plant safely, thus reducing the likelihood of a loss-of-coolant accident.

This criterion is intended to ensure that Technical Specifications control those instruments specifically installed to detect excessive reactor coolant system leakage. This criterion should not, however, be interpreted to include instrumentation to detect precursors to reactor coolant pressure boundary leakage or instrumentation to identify the source of actual leakage (e.g., loose parts monitor, seismic instrumentation, valve position indicators).

Criterion 2 A process variable, design feature, or operating restriction that is an initial condition of a design basis accident (DBA) or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Discussion of Criterion 2

Another basic concept in the adequate protection of the public health and safety is that the plant shall be operated within the bounds of the initial conditions assumed in the existing design basis accident and transient analyses and that the plant will be operated to preclude unanalyzed transients and accidents. These analyses consist of postulated events, (ANSI N18.2) (or equivalent) that either assume the failure of or present a challenge to the integrity of a fission product barrier.

As used in Criterion 2, process variables are only those parameters for which specific values or ranges of values have been chosen as reference bounds in the design basis accident or transient analyses and which are monitored and controlled during power operation such that process values remain within the analysis bounds. Process variables captured by Criterion 2 are not, however, limited to only those directly monitored and controlled from the control room.

These could also include other features or characteristics that are specifically assumed in Design Basis Accident and Transient analyses even if they cannot be directly observed in the control room (e.g., moderator temperature coefficient and hot channel factors).

The purpose of this criterion is to capture those process variables that have initial values assumed in the design basis accident and transient analyses, and which are monitored and controlled during power operation. As long as these variables are maintained within the established values, risk to the public safety is presumed to be acceptably low. This criterion also includes active design features (e.g., high pressure/low pressure system valves and interlocks) and operating restrictions (pressure/temperature limits) needed to preclude unanalyzed accidents and transients.

Criterion 3 A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Discussion of Criterion 3

A third concept in the adequate protection of the public health and safety is that in the event that a postulated design basis accident or transient should occur, structures, systems, and components are available to function or to actuate in order to mitigate the consequences of the design basis accident or transient. Safety sequence analyses or their equivalent have been performed in recent years and provide a method of presenting the plant response to an accident. These can be used to define the primary success paths.

A safety sequence analysis is a systematic examination of the actions required to mitigate the consequences of events considered in the plant's design basis accident and transient analyses, as presented in Chapters 6 and 15 of the plant's Final Safety Analysis Report (or equivalent chapters). Such a safety sequence analysis considers all applicable events, whether explicitly or implicitly presented. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria), so that the plant response to design basis accidents and transients limits the consequences of these events to within the appropriate acceptance criteria.

It is the intent of this criterion to capture into Technical Specifications only those structures, systems, and components that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path for a particular mode of operation does not include backup and diverse equipment (e.g., rod withdrawal block which is a backup to the average power range monitor high flux trip in the startup mode, safety valves which are backup to low temperature overpressure relief valves during cold shutdown).

Criterion 4 A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

<u>Discussion of Criterion 3</u>

It is the Commission's policy that licensees retain in their Technical Specifications LCOs, Action statements and Surveillance Requirements for the following systems (as applicable), which operating experience and PRA have generally shown to be significant to public health and safety and any other structures, systems, or components that meet this criterion:

- Reactor Core Isolation Cooling/Isolation Condenser,
- Residual Heat Removal,
- Standby Liquid Control, and
- Recirculation Pump Trip.

The Commission recognizes that other structures, systems, or components may meet this criterion. Plant and design-specific PRA's have yielded valuable insight to unique plant vulnerabilities not fully recognized in the safety analysis report Design Basis Accident or Transient analyses. It is the intent of this criterion that those requirements that PRA or operating experience exposes as significant to public health and safety, consistent with the Commission's Safety Goal and Severe Accident Policies, be retained or included in Technical Specifications.

> The Commission expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant specific PRA or risk survey and any available literature on risk insights and PRAs. This material should be employed to strengthen the technical bases for those requirements that remain in Technical Specifications, when applicable, and to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk.
>
> Similarly, the NRC staff will also employ risk insights and PRAs in evaluating Technical Specifications related submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements.

3. PRA INSIGHTS

Introduction and Objects

Reference 3 includes a statement that the NRC expects licensees to utilize any plant specific PRA or risk survey and any available literature on risk insights and PRAs to strengthen the technical bases for these requirements that remain in Technical Specifications and to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk.

Those Technical Specifications proposed as being relocated to other plant-controlled documents will be maintained under programs subject to the 10 CFR 50.59 review process or similar processes subject to NRC review. These Relocated Specifications have been compared to a variety of PRA material with two purposes: 1) to identify if a Specification component or topic is addressed by PRA; and 2) if addressed, to judge if the Relocated Specification component or topic is risk-important. The intent of the PRA review was to provide an additional screen to the deterministic criteria. This review was accomplished in the generic Westinghouse Owners Group submittal WCAP-11618 and Addendum 1 to WCAP-11618 (Reference 1). The results of this generic review have been confirmed by FPL for the applicable Turkey Point (PTN) Specifications to be relocated.

Assumptions and Approach

The WCAP-11618 evaluation of the risk impact of the Technical Specifications that are relocation candidates was based on the following:

a. It was assumed that any of the Technical Specifications that were to be relocated would be transferred to other documents subject to control by the utility under the 10 CFR 50.59 process.

b. The risk criteria used in determining the disposition of a Technical Specification were the following:

  1. If the Technical Specification contained constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk, it should be retained;

  2. If the Technical Specification included items involved in one of these dominant sequences but had an insignificant impact on the probability or severity of that sequence, it was proposed to be relocated to another controlled document; and

  3. If the Technical Specification was not involved in risk dominant sequences, it was proposed to be relocated to another controlled document.

c. The measures related to risk used in this evaluation were core damage frequency and off-site health effects. These measures were consistent with the Final Policy Statement on Technical Specifications and the Safety Goal and Severe Accident Policy Statements.

d. The criteria used to determine if a sequence was risk dominant was the following:

   For core damage, any sequence whose frequency was commonly found to be greater than 1 X 10^{-6} per reactor year was maintained as a possible dominant sequence as a conservative first cut. This was roughly 2% of the total core damage frequency of 5 X 10^{-5} for typical PRAs. Each specific sequence identified in the screening of the Technical Specifications was evaluated against the above conservative criterion to determine if it was risk dominant.

   For off-site health effects, any sequence whose frequency of serious radioactive release was commonly found to be greater than 1 x 10^{-7} per reactor year was considered to be a dominant risk sequence for the purposes of WCAP-11618. This criterion was in agreement with the NRC position in the Safety Goal Policy for a goal of 1 X 10^{-6} for a total frequency of severe off-site release, and no greater than 1 X 10^{-7} for an individual sequence.

e. Included in Section 4.0 of WCAP-11618 were two tables (Tables 3 and 4) which contained representative sequences for all identified types of initiating events considered in formal risk assessments for two types of reference plants. Table 3 was representative of a plant with a large dry containment and Table 4 contained the dominant accident sequences for a plant with a subatmospheric containment. These lists were based on industry PRAs and were reviewed for consistency with NRC sponsored PRA programs. The results were found to be consistent.

   Systems identified in Tables 3 and 4 of Section 4.0 of WCAP-11618 that contributed significantly to risk as defined in Paragraph d above were listed in Tables 3A, 3B, 4A, and 4B of Section 4.0. These identified systems as well as sequences and the risk dominant initiating events from Tables 3 and 4 which were involved in typical dominant core damage and serious release sequences from formal risk assessments were used to screen the requirements of the Technical Specifications reviewed. Those Technical

Specifications whose requirements were relevant to these systems, sequences, and initiating events were further evaluated for risk dominance. The remaining Technical Specifications were evaluated on the basis of risk insights from references listed in Section 4.0, Appendix B of WCAP-11618. If the requirements of a Technical Specification were not found to be modeled in any reference and no significant issues were identified from a review of the risk insights, the conclusion was that it did not contain constraints of prime importance to limiting the likelihood or severity of sequences that are commonly found to dominate risk.

4. RESULTS OF APPLICATION OF SELECTION CRITERIA

The selection criteria from Section 2 were applied to the PTN Technical Specifications. The following Summary Disposition Matrix is a summary of that application indicating which Specifications are being retained or relocated, the criteria for inclusion, if applicable, the NRC results of the criteria application as expressed in the NRC Staff Review of NSSS Vendor Owners Groups Application of The Commission's Interim Policy Statement Criteria to Standard Technical Specifications, Wilgus/Murley letter dated May 9, 1988, and any necessary explanatory notes. Discussions that document the rationale for the relocation of each Specification which failed to meet the selection criteria are provided in Appendix A, except as noted in the Summary Disposition Matrix.

5. REFERENCES

1. WCAP-11618 (and Addendum 1), "Methodically Engineered Restructured and Improved Technical Specifications, MERITS Program – Phase II Task 5, Criteria Application," November 1987.

2. NUREG-1431, "Standard Technical Specification, Westinghouse Plants," Revision 5.0

3. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132)

ATTACHMENT 1

SUMMARY DISPOSITION MATRIX FOR PTN

CURRENT TS	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES
		1.0 Use and Applications		
1.0	**Definitions**	1.1 Definitions	Yes-NA	This section provides definitions for several defined terms used throughout the remainder of the Technical Specifications (TSs). The definitions are provided to identify the meaning of certain terms. As such, direct application of the selection criteria is not appropriate. However, only those definitions for defined terms that remain as a result of application of the selection criteria will remain as definitions in this section of the Improved Technical Specifications (ITS).
NA	NA	1.2 Logical Connectors	NA	
NA	NA	1.3 Completion Times	NA	
NA	NA	1.4 Frequency	NA	
2.0	**Safety Limits and Limiting Safety System Settings**			
2.1	**Safety Limits**			
2.1.1	Reactor Core (coolant core outlet pressure and outlet temperature limits)	2.1.1.3	Yes-NA	Application of Technical Specification selection criteria is not appropriate. However, Safety Limits are included in the ITS as required by 10 CFR 50.36.
2.1.2	Reactor Core (thermal power and axial power imbalance limits)	3.3.1	Yes-NA	Same as above.
2.2	**Limiting Safety System Settings**			
2.2.1	Reactor Protection System Instrumentation Setpoints	3.3.1	Yes-NA	Application of Technical Specification selection criteria is not appropriate. However, the Reactor Protection System (RPS) and Limiting Safety System Settings (LSSS) are included as part of the RPS instrumentation Specification, which is retained since the functions either actuate to mitigate the consequences of DBAs and transients or are retained as directed by the NRC as the functions are part of RPS.

CURRENT TS	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES
3.0	**Limiting Conditions for Operation**			
3.0.1	Operational Mode applicability for LCO requirements	3.0.1	Yes-NA	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operations (LCOs)and Surveillance Requirements (SRs). As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements are retained in the ITS, as modified consistent with NUREG-1431.
3.0.2	Compliance with the Specifications	LCO 3.0.2	Yes-NA	Same as above.
3.0.3	Generic Actions for noncompliance	LCO 3.0.3	Yes-NA	Same as above.
3.0.4	Entry into Operational Mode restrictions	LCO 3.0.4	Yes-NA	Same as above.
3.0.5	LCO and ACTION Application for Combined TS	LCO 3.0.10	Yes-NA	This Specification is being retained in the ITS to clarify the application of Turkey Point Unit 3 and Unit 4 (PTN) Combined TS for TS LCO and Actions.
3.0.6	Action Exceptions	LCO 3.0.5	Yes-NA	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of LCOs and SRs. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements are retained in the ITS, as modified consistent with NUREG-1431.
3.0.7	Support/Supported Systems	LCO 3.0.6	Yes-NA	Same as above
NA		LCO 3.0.7		Test Exceptions
NA		LCO 3.0.8		Snubbers
NA		LCO 3.0.9		Barriers

CURRENT TS	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES
4.0	Surveillance Requirements	4.0 SR Applicability		
4.0.1	Operational Mode applicability for surveillance requirements	SR 3.0.1	Yes-NA	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of LCOs and SRs. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements are retained in the ITS, as modified consistent with NUREG-1431, Revision 5.
4.0.2	Time of performance	SR 3.0.2	Yes-NA	Same as above.
4.0.3	Compliance with surveillance requirements	SR 3.0.3	Yes-NA	Same as above.
4.0.4	Entry into Operational Modes	SR 3.0.4	Yes-NA	Same as above.
4.0.5	Deleted	NA	NA	This requirement was deleted in Current Technical Specification (CTS) Amendments 281 and 275.
4.0.6	Application of SRs with Combined TS	SR 3.0.5	Yes-NA	This Specification is being retained in the ITS to clarify the application of PTN Combined TS for TS SRs.
3/4.1	Reactivity Control Systems	**3.1 Reactivity Control Systems**		
3.1.1.1	Shutdown Margin – Tavg Greater Than 200°F	3.1.1 Shutdown Margin (SDM) 3.1.5 Shutdown Bank Insertion Limits 3.1.6 Control Bank Insertion Limits	Yes-2	
3.1.1.2	SDM Tavg Less Than or Equal to 200°F	3.1.1 SDM	Yes-2	
3.1.1.3	Moderator Temperature Coefficient	3.1.3 Moderator Temperature Coefficient (MTC)	Yes-2	
3.1.1.4	Minimum Temperature for Criticality	3.4.2 RCS Minimum Temperature for Criticality	Yes-2	
3.1.2.1	Flow Path - Shutdown	NA	No	Relocate. See Appendix A.

CURRENT TS	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES
3.1.2.2	Flow Paths - Operating	NA	No	Relocate. See Appendix A.
3.1.2.3	Charging Pumps - Operating	NA	No	Relocate. See Appendix A.
3.1.2.4	Borated Water Source - Shutdown	NA	No	Relocate. See Appendix A.
3.1.2.5	Borated Water Sources - Operating	NA	No	Relocate. See Appendix A.
3.1.3.1	Group Height	3.1.4 Rod Group Alignment Limits	Yes-2	
3.1.3.2	Position Indication Systems - Operating	3.1.7 Rod Position Indication	Yes-2	
3.1.3.3	Position Indication System - Shutdown	NA	No	Relocate. See Appendix A.
3.1.3.4	Rod Drop Time	SR 3.1.4.3 Verify rod drop time…	Yes-2	
3.1.3.5	Shutdown Rod Insertion Limit	3.1.5 Shutdown Bank Insertion Limit	Yes-2	
3.1.3.6	Control Rod Insertion Limits	3.1.6 Control Bank Insertion Limits	Yes-2	
NA	NA	3.1.2 Core Reactivity	NA	Add to PTN ITS
3/4.2	**Power Distribution Limits**	**3.2 Power Distribution Limits**		
3.2.1	Axial Flux Difference	3.2.3 Axial Flux Difference (AFD)	Yes-2	
3.2.2	Heat flux Hot Channel Factor $F_Q^{(Z)}$	3.2.1 Heat Flux Hot Channel Factor (F_Q(Z)	Yes-2	
3.2.3	Nuclear Enthalpy Rise Hot Channel Factor	3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)	Yes-2	
3.2.4	Quadrant Power Tilt Ratio	3.2.4 Quadrant Power Tilt Ratio (QPTR)	Yes-2	
3.2.5	DNB Parameters	3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	Yes-2	
3/4.3	**Instrumentation**	**3.3 Instrumentation**		

CURRENT TS	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES
3.3.1	Reactor Trip System Instrumentation	3.3.1 Reactor Trip System (RTS) Instrumentation	Yes-3	
3.3.2	ESFAS	3.3.2 Engineered Safety Features Actuation System (ESFAS) Instrumentation 3.3.4 Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation 3.3.5 Loss of Power Emergency Diesel Generator (EDG) Start Instrumentation 3.3.6 Containment Ventilation Isolation Instrumentation	Yes-3	
3.3.3.1	Radiation Monitoring Instrumentation	3.3.6 Containment Ventilation Isolation Instrumentation 3.4.15 RCS Leakage Detection Instrumentation	Yes-3	CTS 3.3.2 CTS 3.4.5.1
3.3.3.2	Movable Incore Detectors	NA	No	Relocate. See Appendix A.
3.3.3.3	Accident Monitoring Instrumentation	3.3.3 Post Accident Monitoring (PAM) Instrumentation	Yes-3	
3.4.4	Reactor Coolant System	3.4 Reactor Coolant System		
3.4.1.1	Reactor Coolant Loops and Coolant Circulation – Startup and Power Operation	3.4.4 RCS Loops – MODES 1 and 2	Yes-2	
3.4.1.2	RCS – Hot Standby	3.4.5 RCS Loops – MODE 3	Yes-3	
3.4.1.3	RCS – Hot Shutdown	3.4.6 RCS Loops – MODE 4	Yes-4	
3.4.1.4.1	RCS Cold Shutdown – Loops Filled	3.4.7 RCS Loops – MODE 5, Loops Filled	Yes-4	

APPLICATION OF SELECTION CRITERIA TO THE TURKEY POINT NUCLEAR GENERATING STATION UNIT 3 AND UNIT 4 TECHNICAL SPECIFICATIONS

Appendix A – Justification for Specification Relocation

CURRENT TS	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES
3.4.1.4.2	Cold Shutdown – Loops Not Filled	3.4.8 RCS Loops – MODE 5, Loops Not Filled	Yes-4	
3.4.2.1	[Safety Valves – Shutdown] (MODES 4 and 5)	3.4.10 Pressure Safety Valves	Yes-3	
3.4.2.2	[Safety Valves – Operating] (MODES 1, 2, & 3)	3.4.10 Pressure Safety Valves	Yes-3	
3.4.3	Pressurizer	3.4.9 Pressurizer	Yes-2	
3.4.4	Relief Valves	3.4.11 Pressurizer Power Operated Relief Valves (PORVs) Block Valves	Yes-3	The PORVs are move to the Bases (see Appendix A). The PORV block valves, however, are included in the PTN ITS.
3.4.5	Steam Generator (SG) Tube Integrity	3.4.17 Steam Generator (SG) Tube Integrity	Yes-2	SG Tube Integrity is 3.4.20 in the NUREG
3.4.6.1	RCS Leakage Detection Systems	3.4.15 RCS Leakage Detection Instrumentation	Yes-1	
3.4.6.2	RCS Operational Leakage (except Pressure Isolation Valve (PIV) leakage)	3.4.13 RCS Operational Leakage	Yes-2	
3.4.6.2	RCS Operational Leakage - PIV Leakage	3.4.14 RCS Pressure Isolation Valve (PIV) Leakage	Yes-2	
3.4.7	Deleted	NA	NA	Deleted in CTS
3.4.8	Specific Activity	3.4.16 RCS Specific Activity	Yes-2	
3.4.9.1	RCS Pressure Temperature Limits	3.4.3 RCS Pressure and Temperature (P/T) Limits	Yes-2	
3.4.9.2	Pressurizer Temperature	NA	No	Relocate. See Appendix A.
3.4.9.3	Overpressure Mitigating Systems	3.4.12 Overpressure Mitigating System	Yes-2	
3.4.10	Deleted in CTS	NA	NA	
3.4.11	Reactor Coolant System Vents	NA	No	Relocate. See Appendix A.

CURRENT TS	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES
3/4.5 ECCS		**3.5 Emergency Core Cooling Systems (ECCS)**		
3.5.1	Accumulators	3.5.1 Accumulators	Yes-3	
3.5.2	ECCS Subsystems - T_{avg} ≥ 350 °F	3.5.2 ECCS – Operating	Yes-3	
3.5.3	ECCS Subsystem – T_{avg} < 350 °F	3.5.3 ECCS – Shutdown	Yes-3	
3.5.4	Refueling Water Storage Tank	3.5.4 Refueling Water Storage Tank (RWST)	Yes-3	
3/4.6 Containment Systems		**3.6 Containment Systems**		
3.6.1.1	Containment Integrity	3.6.1 Containment	Yes-3	
3.6.1.2	Containment Leakage	3.6.1 Containment	Yes-3	
3.6.1.3	Containment Air Locks	3.6.2 Containment Air Locks	Yes-3	
3.6.1.4	Containment Internal Pressure	3.6.4 Containment Pressure	Yes-2	
3.6.1.5	Containment Air Temperature	3.6.5 Containment Air Temperature	Yes-2	
3.6.1.6	Containment Structural Integrity	3.6.1 Containment	Yes-3	
3.6.1.7	Containment Ventilation System	3.6.3 Containment Isolation Valves	Yes-3	
3.6.2.1	Containment Spray System	3.6.6 Containment Spray and Cooling Systems	Yes-3	
3.6.2.2	Emergency Containment Cooling System	3.6.6 Containment Spray and Cooling Systems	Yes-3	
3.6.2.3	Recirculation pH Control System	3.6.7 Recirculation pH Control System	Yes-3	
3.6.4	Containment Isolation Valves	3.6.3 Containment Isolation valves	Yes-3	
3/4.7	Plant Systems	3.7 Plant Systems		
3.7.1.1	Safety Valves	3.7.1 Main Steam Safety Valves (MSSVs)	Yes-3	

APPLICATION OF SELECTION CRITERIA TO THE TURKEY POINT NUCLEAR GENERATING STATION UNIT 3 AND UNIT 4 TECHNICAL SPECIFICATIONS

Appendix A – Justification for Specification Relocation

CURRENT TS	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES
3.7.1.2	Auxiliary feedwater System	3.7.5 Auxiliary Feedwater (AFW) System	Yes-3	
3.7.1.3	Condensate Storage Tank	3.7.6 Condensate Storage Tank (CST)	Yes-2/3	
3.7.1.4	Specific Activity	3.7.4 Secondary Specific Activity	Yes-2	Relocate Details (Table) to the TRM). Secondary Specific Activity is 3.7.18 in NUREG.
3.7.1.5	Main Steam Isolation Valves	3.7.2 Main Steam Isolation Valves (MSIVs)	Yes-2	
3.7.1.6	Deleted	NA	NA	Deleted in Amendments 282 and 276
3.7.1.7	Feedwater Isolation	3.7.3 Feedwater Isolation Valves (FIVs) and Feedwater Control Valves (FCVs)	Yes-3	
3.7.2	Component Cooling Water System	3.7.7 Component Cooling Water (CCW) System	Yes-3	
3.7.3	Intake Cooling Water System	3.7.8 Intake Cooling Water (ICW) System (ICWS)	Yes-3	
3.7.4	Ultimate Heat Sink	3.7.9 Ultimate Heat Sink (UHS)	Yes-3	
3.7.5	Control Room Emergency Ventilation System	3.7.10 Control Room Emergency Ventilation System (CREVS) 5.5.9 Ventilation Filter Testing Program	Yes-3	
3.7.5	Control Room Emergency Ventilation System	3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)	Yes-3	
3.7.6	Snubbers	NA	Yes	Move to LCO 3.0.8
3.7.7	Sealed Source Contamination	NA	No	Relocate. See Appendix A.
3.9.11	Water Level Storage Pool	3.7.12 Fuel Storage Pool Water Level	YES-3	Fuel Storage Pool Water Level is 3.7.15 in NUREG.
3.9.14	Spent Fuel Storage	3.7.13 Fuel Storage Pool Boron Concentration	YES-3	Fuel Storage Pool Boron Concentration is 3.7.16 in NUREG
3.9.14	Spent Fuel Storage	3.7.14 Spent Fuel Storage	YES-3	Spent Fuel Storage is 3.7.17 in NUREG

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TECHNICAL SPECIFICATIONS

Appendix A – Justification for Specification Relocation

CURRENT TS	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES
3/4.8	Electrical Power Systems	3.8 Electrical Power Systems		
3.8.1.1	AC Sources – Operating	3.8.1 AC Sources – Operating	Yes-3	
3.8.1.1	AC Sources – Operating	3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air	Yes-3	
3.8.1.2	AC Sources – Shutdown	3.8.2 AC Sources – Shutdown	Yes-3	
3.8.2.1	DC Sources – Operating	3.8.4 DC Sources – Operating	Yes-3	
3.8.2.1	DC Sources – Operating	3.8.6 Battery Parameters	Yes-3	
3.8.2.2	DC Sources – Shutdown	3.8.5 DC Sources – Shutdown	Yes-3	
3.8.3.1	Onsite Power Distribution – Operating	3.8.9 Distribution Systems – Operating	Yes-3	
3.8.3.2	Onsite Power Distribution – Shutdown	3.8.10 Distribution Systems – Shutdown	Yes-3	
NA	NA	3.8.7 Inverters – Operating	NA	Add Inverters to PTN ITS
NA	NA	3.8.8 Inverters – Shutdown	NA	Add Inverters to PTN ITS
3/4.9	Refueling Operations	3.9 Refueling Operations		
3.9.1	Boron concentration	3.9.1 Boron Concentration	Yes-2	
3.9.2	Instrumentation	3.9.3 Nuclear Instrumentation	Yes-3	
3.9.3	Decay Time	NA	NA	LA to the TRM
3.9.4	Containment Building Penetrations	3.9.4 Containment Penetrations	Yes-3	
3.9.5	Deleted	NA	NA	Deleted in CTS
3.9.6	Deleted	NA	NA	Deleted in CTS
3.9.7	Deleted	NA	NA	Deleted in CTS

CURRENT TS	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES
3.9.8.1	Residual Heat Removal and Coolant Circulation - High Water Level	3.9.5 Residual Heat Removal and Coolant Circulation - High Water Level	Yes-3	
3.9.8.2	Refueling Operations – Low Water Level	3.9.6 Residual Heat Removal and Coolant Circulation - Low Water Level	Yes-3	
3.9.9	Containment Ventilation Isolation System	3.9.4 Containment Penetrations	Yes-3	
3.9.10	Refueling Cavity Water Level	3.9.2 Refueling Cavity Water Level	Yes-3	Refueling Cavity Water Level is 3.9.7 in the NUREG.
3.9.11	Water Level Storage Pool	3.7.12 Fuel Storage Pool Water Level	Yes-3	
3.9.12	Deleted	NA	NA	Deleted in CTS
3.9.13	Radiation Monitoring	NA	No	Relocate. See Appendix A.
3.9.14	Spent Fuel Storage	3.7.14 Spent Fuel Storage	Yes-3	
3/4.10	Special Test Exceptions			
3.10.1	Special Test Exceptions Shutdown Margin	Deleted	No	Delete - See Discussion of Changes for CTS 3/4.10.1.
3.10.2	Special Test Exceptions Group Height, Insertion, and Power Distribution Limits	Deleted	No	Delete - Discussion of Changes for CTS 3/4.10.2.
3.10.3	Special Test Exceptions Physics Tests	3.1.8 PHYSICS TESTS Exceptions – MODE 2	Yes-3	Although this Specification does not meet any Technical Specification selection criteria, it has been retained to provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs.
3.10.4	NOT USED	NA	NA	
3.10.5	Special Test Exceptions Position Indication System – Shutdown	Deleted	No	See Discussion of Changes for CTS 3.10.5.
5.0	DESIGN FEATURES	4.0	Yes-NA	
6.0	Administrative Controls	5.0	Yes-NA	

APPENDIX A

JUISTIFICATION FOR SPECIFICATION RELOCATION

CTS 3.1.2.1, FLOW PATHS – SHUTDOWN

CTS 3.1.2.2, FLOW PATHS – OPERATING

CTS 3.1.2.3, CHARGING PUMPS – OPERATING

CTS 3.1.2.4, BORATED WATER SOURCE – SHUTDOWN

CTS 3.1.2.5, BORATED WATER SOURCE – SHUTDOWN

<u>Discussion:</u>

Turkey Point Nuclear Generating Station (PTN) Current Technical Specifications (CTS) 3.1.2.1 provides the requirements for the minimum boron injection flow paths (one) during shutdown (Modes 5 and 6). CTS 3.1.2.2 provides the requirements for the minimum boron injection flow paths (2) during operation Modes 1 - 4. CTS 3.1.2.3 provides the requirement to have two charging pumps available during Modes 1 – 4 as the motive means to transfer the boron inventory to the Reactor Coolant System (RCS) during normal operation. CTS 3.1.2.4 requires, as a minimum, one borated water source (Boric Acid Storage System or Refueling Water Storage Tank) to be operable during Modes 5 and 6. CTS 3.1.2.5 requires both borated water sources (Boric Acid Storage System and Refueling Water Storage Tank) to be operable during Modes 1 - 4.

The components associated with the boration system Technical Specifications (TSs) provide the means to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the shutdown margin (SDM) during normal operations. To accomplish this functional requirement, the current boration system TSs require a source(s) of borated water, one or more flow paths to inject borated water into the RCS and Charging Pumps to provide the necessary charging head.

The boration systems are not assumed to be operable to mitigate the consequences of a Design Basis Accident (DBA) or transient. In the case of a malfunction of a component in the boration systems which causes a boron dilution event, the automatic response, or that required by the operator, is to close the appropriate valves in the reactor makeup system. The automatic plant response to a boron dilution event also includes automatic control rod assembly movement and reactor trip features to ensure shutdown margin is maintained. The boration capabilities of the boration systems are not assumed to mitigate the boron dilution event.

<u>Comparison to Selection Criteria:</u>

1. The boration systems do not constitute an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The boration systems are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier. This TS specifies limits on process variables consistent with the structural analysis results. These limits, however, do not reflect initial condition assumptions in the DBA.

3. The boration systems are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The function of injecting borated water to maintain shutdown margin is not risk significant. Operational experience has shown that the boration management system is not a constraint of prime importance in the mitigation of any accident or transient that results in challenging public health and safety. Therefore, the RCS boration management system functions to control boron concentration and maintain shutdown margin do not represent structures, systems, or components which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Conclusion:

Since the selection criteria have not been satisfied, the boration system Limiting Conditions for Operation (LCOs) and Surveillances, may be relocated to licensee-controlled documents outside the TSs. Operability requirements for Emergency Boration and Residual Heat Removal (RHR) pumps that support core RHR capability and minimum boration requirements during plant shutdown, are retained in separate TSs.

CTS 3.1.3.3, POSITION INDICATION SYSTEM – SHUTDOWN

Discussion:

CTS 3.1.3.3 provides the requirements for the group step counter demand position indicator to be operable and capable of determining within ± 2 steps the demand position for each shutdown and control rod not fully inserted in Modes 3, 4 and 5 with the reactor trip breakers in the closed position.

Rod position indication ensure operability of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. In Modes 3, 4, and 5, SDM is required per LCO 3.1.1 which references the Core Operating Limits Report (COLR). The COLR requires sufficient reactivity margin to ensure fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. This sufficient reactivity margin takes into account rod positions with the single rod cluster assembly of the highest reactivity worth fully withdrawn. In the shutdown Modes, the operability of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

Comparison to Selection Criteria:

1. Control Rod Position Indications in Modes 3, 4, and 5 do not constitute an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. Control Rod Position Indications in Modes 3, 4, and 5 are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier. This TS specifies limits on process variables consistent with the structural analysis results. These limits, however, do not reflect initial condition assumptions in the DBA.

3. Control Rod Position Indications in Modes 3, 4, and 5 are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. Control Rod Position Indications in Modes 3, 4, and 5 were found to be non-significant risk contributor to core damage frequency and offsite releases. These indications are not structures, systems, or components that operating experience or probabilistic safety assessment has shown to be significant to the public health and safety.

Conclusion:

Since the selection criteria have not been satisfied, the Modes 3, 4, and 5 Control Rod Position indications in LCO and Surveillances may be relocated to licensee-controlled documents outside the TSs. Position Indication requirements in Modes 1 and 2 are required by LCO 3.1.7 to ensure the initial conditions of the Safety Analyses are maintained.

CTS 3.3.3.2, MOVABLE INCORE DETECTORS

Discussion:

CTS 3.3.3.2 provides the requirements for Movable Incore Detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The operability of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

This requirement and the associated Surveillance Requirements (SRs) bear no relation to the conditions or limitations that are necessary to ensure safe reactor operation. While the incores can provide monitoring capability, the detectors are mainly utilized to recalibrate the excore detectors and do not provide input to any trip function. The excores are credited and utilized to provide input to the Reactor Trip System (RTS).

Comparison to Selection Criteria:

1. Movable Incore Detectors do not constitute an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. Movable Incore Detectors are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier. This TS specifies limits on process variables consistent with the structural analysis results. These limits, however, do not reflect initial condition assumptions in the DBA.

3. Movable Incore Detectors are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. Movable Incore Detectors were found to be non-significant risk contributor to core damage frequency and offsite releases. These indications are not structures, systems, or components that operating experience or probabilistic safety assessment has shown to be significant to the public health and safety.

Conclusion:

Since the selection criteria have not been satisfied, the Movable Incore Detectors LCO and Surveillances, may be relocated to licensee-controlled documents outside the TSs. The movable incores do not provide input to any trip system. The excores are credited and utilized in the RPS.

CTS 3.4.9.2, PRESSURIZER TEMPERATURE

Discussion:

CTS 3.4.9.2 provides for the maximum cooldown and heatup temperatures per hour (shall not exceed 200 °F/hr and 100 °F/hr, respectively) for the Pressurizer and the maximum spray water temperature differential (> 320 °F). The limits meet the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G. These limitations are consistent with structural analysis results. However, these limits are not initial condition assumptions of a DBA or transient. These limits represent operating restrictions and Criterion 2 includes operating restrictions. However, it should be noted that in the Final Policy Station, the Criterion 2 discussion specified only those operating restrictions required to preclude unanalyzed accidents and transients be included in TSs. This Specification does not meet the criteria for retention in the Improved Technical Specifications (ITS).

Comparison to Selection Criteria:

1. The pressurizer temperature limits are not installed instrumentation that are used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The pressurizer temperature limits are not a process variable, design feature, or operating restriction that is an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The pressurizer temperature limits are not a structure, system, or component that is part of primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The pressurizer temperature limits are not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. The pressurizer temperature limits were found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the selection criteria have not been satisfied, the Pressurizer Temperature limits LCO, and Surveillances may be relocated to licensee-controlled documents outside the TSs. The Pressurizer temperature limits Specification will be relocated to the Technical Requirements Manual (TRM). Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the Specification did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

CTS 3.7.7, SEALED SOURCE CONTAMINATION

Discussion:

CTS 3.7.7 provides the requirements that each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, are based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

This requirement and the associated SRs bear no relation to the conditions or limitations that are necessary to ensure safe reactor operation.

Comparison to Selection Criteria:

1. Sealed Source Contamination limitations do not constitute an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. Sealed Source Contamination limitations are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier. This TS specifies limits on process variables consistent with the structural analysis results. These limits, however, do not reflect initial condition assumptions in the DBA.

3. Sealed Source Contamination limitations are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. Sealed Source Contamination limitations were found to be non-significant risk contributor to core damage frequency and offsite releases. These indications are not structures, systems, or components that operating experience or probabilistic safety assessment has shown to be significant to the public health and safety.

Conclusion:

Since the selection criteria have not been satisfied, the Sealed Source Contamination LCO and Surveillances, may be relocated to licensee-controlled documents outside the TSs. Requirements associated with the sealed sources are governed by 10 CFR Part 70. Compliance with applicable portions of 10 CFR Part 70 is required by the operating licenses of PTN Units 3 and 4.

CTS 3.9.13, Radiation Monitoring

Discussion:

CTS 3.9.13 provides a high radiation signal from the Radiation Monitors to the Containment Ventilation Isolation System during movement of irradiated fuel within containment. The Instrumentation required by these TSs minimize the release of airborne radiation to the outside atmosphere during a fuel handling accident. However, the fuel handling accident for PTN does not credit automatic closure of containment during the fuel handling accident. The release is assumed to leak to the environment for two hours without filtration. As an additional conservatism, all the fuel rods in a single assembly are assumed to be damaged.

The Radiation Monitors required by TS 3.9.13 can be relocated out of TSs because the structures, systems, or components are not credited during the Fuel Handling Accident.

Comparison to Selection Criteria:

1. The Containment Radiation Monitors do not constitute an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. The Containment Radiation Monitoring instrumentation are not credited in the Fuel Handling Accident.

2. The Containment Radiation Monitors are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier. This TS specifies limits on process variables consistent with the structural analysis results. These limits, however, do not reflect initial condition assumptions in the DBA.

3. The Containment Radiation Monitors are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The Containment Ventilation Isolation System and the associated Radiation Monitors that operate to automatically isolate containment upon detecting high radiation are not credited in the Fuel Handling Accident to isolate containment. The Fuel Handling Accident assumes no automatic isolation and the release continues over a two-hour period.

4. The Containment Radiation Monitors were found to be non-significant risk contributor to core damage frequency and offsite releases. These indications are not structures, systems, or components that operating experience or probabilistic safety assessment has shown to be significant to the public health and safety. These Systems are not credited to operate during a Fuel Handling Accident.

Conclusion:

Since the selection criteria have not been satisfied, the Containment Radiation Monitors LCO and Surveillances may be relocated to licensee-controlled documents outside the TSs.

Package Supporting Information

None