ENCLOSURE 2

VOLUME 1

SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

APPLICATION OF SELECTION CRITERIA TO THE SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2 TECHNICAL SPECIFICATIONS

APPLICATION OF SELECTION CRITERIA TO THE
SEQUOYAH NUCLEAR PLANT
UNIT 1 AND UNIT 2
TECHNICAL SPECIFICATIONS

Revision 0

**APPLICATION OF SELECTION CRITERIA
TO THE SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2
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**APPLICATION OF SELECTION CRITERIA
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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 Sequoyah Nuclear Plant (SQN) Unit 1 and Unit 2. The Tennessee Valley Authority (hereinafter TVA) 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, 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 4.0 "Standard Technical Specifications, Westinghouse Plants" (Reference 2) and applied the criteria to each of the current SQN 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 SQN.

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2. SELECTION CRITERIA

TVA 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 TVA, and are discussed in the next section of this report. The selection criteria and discussion provided in Reference 3 are as follows:

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

Discussion of Criterion 1: 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, analyzed in the Final Safety Analysis Report (FSAR), for which a structure, system, or component must meet specified functional goals. These analyses are contained in Chapters 6 and 15 of the FSAR (or equivalent chapters) and are identified as Condition II, III, or IV events

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(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:

Discussion of Criterion 4: 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 Objectives

Reference 3 includes a statement that 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. 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 TVA for the applicable SQN Specifications to be relocated.

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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 1×10^{-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 5×10^{-5} for typical PRAs. Each specific sequence identified in the screening of the Technical

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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 1×10^{-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 1×10^{-6} for a total frequency of severe off-site release, and no greater than 1×10^{-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.

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4. RESULTS OF APPLICATION OF SELECTION CRITERIA

The selection criteria from Section 2 were applied to the SQN 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.

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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 Specifications, Westinghouse Plants," Revision 4.0, April 2001.

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

ATTACHMENT 1

SUMMARY DISPOSITION MATRIX
FOR
SEQUOYAH NUCLEAR PLANT
UNIT 1 AND UNIT 2

SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
1.0 1.3 1.7 1.30	**DEFINITIONS**	**1.1** 5.5.1 3.6.3 3.6.1/3.6.3 3.6.7/3.6.10	YES	This section provides definitions for several defined terms used throughout the remainder of Technical Specifications. They are provided to improve the meaning of certain terms. As such, direct application of the Technical Specification 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 Technical Specifications.
2.0	**SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS**	**2.0**		
2.1	***Safety Limits***	***2.1***		
2.1.1	Reactor Core-The combination of Thermal Power, Pressurizer Pressure and Highest Operating Loop Coolant Tavg Shall not exceed the limits of the COLR	2.1.1	YES	Application of Technical Specification selection criteria is not appropriate. However, Safety Limits will be included in Technical Specifications as required by 10 CFR 50.36.

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
2.1.2	Reactor Coolant System Pressure- Reactor Coolant System Pressure shall not exceed 2735	2.1.2	YES	Application of Technical Specification selection criteria is not appropriate. However, Safety Limits will be included in Technical Specifications as required by 10 CFR 50.36.
2.2	***Limiting Safety System Settings***			
2.2.1	Reactor Protection System Setpoints	3.3.1	YES-3	The RTS LSSS have been included as part of the RPS instrumentation Specification, which has been retained since the Functions either actuate to mitigate consequences of design basis accidents and transients or are retained as directed by the NRC as the Functions are part of the RTS.

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.0	**LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS - APPLICABILITY**	**3.0**		
3.0.1	Operational Modes	LCO 3.0.1	YES	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operation and Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements of 3.0/4.0 will be retained in Technical Specifications, as modified consistent with NUREG-1431, Revision 4.
3.0.2	Noncompliance	LCO 3.0.2	YES	Same as above.
3.0.3	Generic Actions	LCO 3.0.3	YES	Same as above.
3.0.4	Entry into Operational Modes	LCO 3.0.4	YES	Same as above.
3.0.5	Operability Exception	3.8.1	YES	The application of Technical Specification selection criteria is not appropriate. However, this exception to the definition of OPERABILITY has been included as part of the Required Actions in ITS 3.8.1.

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3.0.6	Actions Exceptions	LCO 3.0.5	YES	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operation and Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements of 3.0/4.0 will be retained in Technical Specifications, as modified consistent with NUREG-1431, Revision 4.
3.0.7	Snubbers	LCO 3.0.8	YES	
4.0.1	Operational Modes	SR 3.0.1	YES	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operation and Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements of 3.0/4.0 will be retained in Technical Specifications, as modified consistent with NUREG-1431, Revision 4.

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
4.0.2	Time of Performance	SR 3.0.2	YES	Same as above.
4.0.3	Noncompliance	SR 3.0.3	YES	Same as above.
4.0.4	Entry into Operational Modes	SR 3.0.4	YES	Same as above.
4.0.5	ASME Code Class 1, 2, and 3 Components	5.5.5 5.5.6	YES	This Specification is actually a Surveillance Requirement which has been retained in the Administrative Controls programs for Inservice Testing.
3/4.1	**REACTIVITY CONTROL SYSTEMS**	**3.1**		
3/4.1.1	***Boration Control***			
3.1.1.1	SHUTDOWN MARGIN- Tavg Greater Than 200°F	1.0 3.1.1 3.1.2 3.1.4 3.1.6	YES-2	
3.1.1.2	SHUTDOWN MARGIN- Tavg Less Than or Equal to 200°F	1.0 3.1.1 3.1.4	YES-2	
3.1.1.3	Moderator Temperature Coefficient	3.1.3	YES-2	
3.1.1.4	Minimum Temperature for Criticality	3.4.2	YES-2	

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.1.2	***Boration Systems (deleted)***			
3/4.1.3	***Movable Control Assemblies***			
3.1.3.1	Group Height	3.1.4	YES-2	
3.1.3.2	Position Indicator Systems-Operating	3.1.7	YES-2	
3.1.3.4	Rod Drop Time	3.1.4	YES-2	This Specification has been incorporated as a Surveillance Requirement (SR 3.1.4.3) in ITS 3.1.4.
3.1.3.5	Shutdown Rod Insertion Limit	3.1.5	YES-2	
3.1.3.6	Control Rod Insertion Limits	3.1.6	YES-2	
3/4.2	**POWER DISTRIBUTION LIMITS**	**3.2**		
3/4.2.1	***Axial Power Imbalance***			
3.2.1	Axial Flux Difference(AFD)	3.2.3	YES-2	
3/4.2.2	***Heat Flux Hot Channel Factor - $F_Q(X,Y,Z)$***			
3.2.2	Heat Flux Hot Channel Factor - $F_Q(X,Y,Z)$	3.2.1	YES-2	

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.2.3	***Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}$ (X,Y)***			
3.2.3	Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}$(X,Y)	3.2.2	YES-2	
3/4.2.4	***Quadrant Power Tilt Ratio***			
3.2.4	Quadrant Power Tilt Ratio	3.2.4	YES-2	
3/4.2.5	***DNB Parameters***			
3.2.5	DNB Parameters	3.4.1	YES-2	
3/4.3	**INSTRUMENTATION**	**3.3**		
3/4.3.1	***Reactor Trip System Instrumentation***			
3.3.1.1 U1 3.3.1 U2	Reactor Trip System Instrumentation	3.3.1 3.3.2 3.3.9	YES-3	
3/4.3.2	***Engineered Safety Feature Actuation System Instrumentation***			
3.3.2.1 U1 3.2.1 U2	Engineered Safety Feature Actuation System Instrumentation	3.3.2 3.3.6	YES-3	

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.3.3	***Monitoring Instrumentation***			
3.3.3.1	Radiation Monitoring Instrumentation	3.3.6 3.3.7 3.3.8	YES-3	
Instrument 1	Area Monitors			
Instrument 1.a	Fuel Storage Pool Area Emergency Ventilation System Actuation	3.3.8	YES-3	
Instrument 2	Process Monitors			
Instrument 2.a	Containment Purge Air	3.3.6	YES-3	
Instrument 2.b.ii	Containment Particulate Activity RCS Leakage Detection	3.4.15	YES-1	
Instrument 2.c	Control Room Isolation	3.3.7	YES-3	
3.3.3.5	Remote Shutdown Instrumentation	3.3.4	YES-4	
3.3.3.7	Accident Monitoring Instrumentation	3.3.3 5.6.5	YES-3	
3.3.3.10	Explosive Gas Monitoring Instrumentation	Relocated	NO	See Appendix A, Page 1.
3.3.3.11	Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation	3.3.5	YES-3	

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.4	**REACTOR COOLANT SYSTEM**	**3.4**		
3/4.4.1	***Reactor Coolant Loops and Coolant Circulation***			
3.4.1.1	Startup and Power Operation	3.4.4	YES-2	
3.4.1.2	Hot Standby	3.4.5	YES-3	
3.4.1.3	Shutdown	3.4.6	YES-4	
3.4.1.4	Cold Shutdown	3.4.7 3.4.8	YES-4	
3/4.4.3	***Safety and Relief Valve - Operating***			
3.4.3.1	Safety Valves-Operating	3.4.10	YES-3	
3.4.3.2	Relief Valves -Operating	3.4.11	YES-3	
3/4.4.4	***Pressurizer***			
3.4.4	Pressurizer	3.4.9	YES-2	
3/4.4.5	***Steam Generators***			
3.4.5	Steam Generators	3.4.17 5.5.9	YES-2	

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.4.6	***Reactor Coolant System Leakage***			
3.4.6.1	Leakage Detection Instrumentation	3.4.15	YES-1	
3.4.6.2	Operational Leakage	3.4.13	YES-2	
3.4.6.3	Reactor Coolant System Pressure isolation Valve Leakage	3.4.14	YES-2	
3/4.4.8	***Specific Activity***			
3.4.8	Specific Activity	3.4.16	YES-2	
3/4.4.9	***RCS Pressure and Temperature (PT)Limits***			
3.4.9.1	RCS Pressure and Temperature (PT)Limits	3.4.3	YES-2	
3/4.4.12	***Low Temperature Over Pressure Protection (LTOP) System***			
3.4.12	Low Temperature Over Pressure Protection (LTOP) System	3.4.12	YES-2	

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.5	**EMERGENCY CORE COOLING SYSTEMS(ECCS)**	3.5		
3/4.5.1	***Accumulators***			
3.5.1	Cold Leg Injection Accumulators	3.5.1	YES-3	
3/4.5.2	***ECCS Subsystems - Operating***			
3.5.2	ECCS Subsystems - Operating	3.5.2	YES-3	
3/4.5.3	***ECCS Subsystems - Shutdown***			
3.5.3	ECCS Subsystems - Shutdown	3.5.3	YES-3	
3/4.5.5	***Refueling Water Storage Tank***			
3.5.5	Refueling Water Storage Tank	3.5.4	YES-3	
3/4.5.6	***Seal Injection Flow***			
3.5.6	Seal Injection Flow	3.5.5	YES-2	

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.6	**CONTAINMENT SYSTEMS**	**3.6**		
3/4.6.1	***Primary Containment***			
3.6.1.1	Containment Integrity	3.6.1 3.6.2	YES-3	
3.6.1.3	Containment Air Locks	3.6.2	YES-3	
3.6.1.4	Internal Pressure	3.6.4	YES-2	
3.6.1.5	Air Temperature	3.6.5	YES-2	
3.6.1.6	Containment Vessel Structural Integrity	3.6.1	YES-3	Containment vessel structural integrity is being retained as a Surveillance Requirement (SR 3.6.1.1) in ITS 3.6.1.
3.6.1.7	Shield Building Structural Integrity	3.6.7	YES-3	
3.6.1.8	Emergency Gas Treatment System-EGTS-Clean Up Subsystems	3.6.10 5.5.9	YES-3	
3/4.6.2	***Depressurization and Cooling Systems***			
3.6.2.1	Containment Spray SubSystems	3.6.6	YES-3	
3.6.2.2	Lower Containment Vent Coolers	Relocated	No	See Appendix A, Page 2.

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.6.3	***Containment Isolation Valves***	***3.6.3***	***YES-3***	
3.6.3	Containment Isolation Valves	3.6.3	YES-3	
3/4.6.4	***Combustible Gas Control***			
3.6.4.3	Hydrogen Mitigation System	3.6.8	YES-4	
3/4.6.5	***Ice Condenser***			
3.6.5.1	Ice Bed	3.6.12	YES-3	
3.6.5.3	Ice Condenser Doors	3.6.13	YES-3	
3.6.5.5	Divider Barrier Personnel Access Doors and Equipment hatches	3.6.14	YES-3	
3.6.5.6	Containment Air Return Fans	3.6.11	YES-3	
3.6.5.7	Floor Drains	3.6.15	YES-3	
3.6.5.8	Refueling Canal Drains	3.6.15	YES-3	
3.6.5.9	Divider Barrier Seals	3.6.14	YES-3	
3/4.6.6	***Vacuum Relief Lines***			
3.6.6	Vacuum Relief lines	3.6.9	YES-3	

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.7	**PLANT SYSTEMS**	**3.7**		
3/4.7.1	***Turbine Cycle***			
3.7.1.1	Safety Valves	3.7.1	YES-3	
3.7.1.2	Auxiliary Feedwater (AFW) System	3.7.5	YES-3	
3.7.1.3	Condensate Storage System	3.7.6	YES-2, 3	
3.7.1.4	Activity	3.7.16	YES-2	
3.7.1.5	Main Steam Line Isolation Valves	3.7.2	YES-3	
3.7.1.6	Main Feedwater Isolation, Regulation and Bypass Valves	3.7.3	YES-3	
3/4.7.3	***Component Cooling Water System***			
3.7.3	Component Cooling Water System	3.7.7	YES-3	
3/4.7.4	***Service Water System***			
3.7.4	Service Water System	3.7.8	YES-3	
3/4.7.5	***Ultimate Heat Sink***			
3.7.5	Ultimate Heat Sink	3.7.9	YES-3	

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.7.7	***Control Room Emergency Ventilation System***			
3.7.7	Control Room Emergency Ventilation System	3.7.10 5.5.9	YES-3	
3/4.7.8	***Auxiliary Building Gas Treatment System***			
3.7.8	Auxiliary Building Gas Treatment System	3.7.12 5.5.9	YES-3	
3/4.7.13	***Spent Fuel Pool Minimum Boron Concentration***			
3.7.13	Spent Fuel Pool Minimum Boron Concentration	3.7.14	YES-2	
3/4.7.14	***Cask Pit Pool Minimum Boron Concentration***			
3.7.14	Cask Pit Pool Minimum Boron Concentration	3.7.17	YES-2	
3/4.7.15	***Control Room Air Conditioning System***			
3.7.15	Control Room Air Conditioning System	3.7.11	YES-3	

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.8	**ELECTRICAL POWER SYSTEM**	**3.8**		
3/4.8.1	***A.C. Sources***			
3.8.1.1	Operating	3.8.1 3.8.3 3.8.4 3.8.6 3.8.9	YES-3	
3.8.1.2	Shutdown	3.8.2 3.8.3 3.8.5 3.8.6 3.8.10	YES-3	
3/4.8.2	***Onsite Power Distribution Systems***			
3.8.2.1	A.C. Distribution - Operating	3.8.7 3.8.9	YES-3	
3.8.2.2	A.C. Distribution - Shutdown	3.8.8 3.8.10	YES-3	
3.8.2.3	D.C. Distribution - Operating	3.8.4 3.8.6 3.8.9	YES-3	

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3.8.2.4	D.C. Distribution - Shutdown	3.8.5 3.8.6 3.8.10	YES-3	
3/4.9	**REFUELING OPERATIONS**	**3.9**		
3/4.9.1	***Boron Concentration***			
3.9.1	Boron Concentration	3.9.1 3.9.2	YES-2	
3/4.9.2	***Instrumentation***			
3.9.2	Instrumentation	3.9.3	YES-3	
3/4.9.3	***Decay Time***			
3/4.9.3	Decay Time	Deleted	NO	See technical change discussion in Enclosure 2, Volume 14, Discussion of Changes for CTS 3/4.9.3.
3/4.9.4	***Containment Building Penetrations***			
3.9.4	Containment Building Penetrations	3.9.4	YES-3	

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.9.8	***Residual Heat Removal and Coolant Circulation***			
3.9.8.1	All Water Levels	3.9.5 3.9.6	YES-4	
3.9.8.2	Low Water Level	3.9.6	YES-4	
3/4.9.9	***Containment Ventilation Isolation System***			
3.9.9	Containment Ventilation Isolation System	3.9.4	YES-3	
3/4.9.10	***Water Level - Reactor Vessel***			
3.9.10	Water Level - Reactor Vessel	3.9.7	YES-2	
3/4.9.11	***Storage Pool Water Level***			
3.9.11	Storage Pool Water Level	3.7.13	YES-2, 3	
3/4.9.12	***Auxiliary Building Gas Treatment System***			
3.9.12	Auxiliary Building Gas Treatment System	3.7.12 5.5.9	YES-3	

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.10	**SPECIAL TEST EXCEPTIONS**			
3/4.10.1	***Shutdown Margin***			
3.10.1	Shutdown Margin	Deleted	NO	See technical change discussion in Enclosure 2, Volume 6, Discussion of Changes for CTS 3/4.10.1.
3/4.10.2	***Group Height, Insertion and Power Distribution Limits***			
3/4.10.2	Group Height, Insertion and Power Distribution Limits	Deleted	NO	See technical change discussion in Enclosure 2, Volume 6, Discussion of Changes for CTS 3/4.10.2.
3/4.10.3	***Physics Tests***			
3.10.3	Physics Tests	3.1.8	YES	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.

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3/4.10.4	***Reactor Coolant Loops***			
3.10.4	Reactor Coolant Loops	Deleted	NO	See technical change discussion in Enclosure 2, Volume 6, Discussion of Changes for CTS 3/4.10.4.
3/4.11	**RADIOACTIVE EFFLUENTS**	**NA**		
3/4.11.1	***Liquid Effluents***			
3.11.1.4	Liquid Holdup Tanks	5.5.10	YES	Although this Specification does not meet any Technical Specification selection criteria, it has been retained in accordance with the NRC letter from W. T. Russell to the industry ITS Chairpersons, dated October 25, 1993.
3/4.11.2	***Gaseous Effluents***			
3.11.2.5	Explosive Gas Mixture	5.5.10	YES	Although this Specification does not meet any Technical Specification selection criteria, it has been retained in accordance with the NRC letter from W. T. Russell to the industry ITS Chairpersons, dated October 25, 1993.

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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SUMMARY DISPOSITION MATRIX FOR SEQUOYAH NUCLEAR PLANT UNIT 1 AND UNIT 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(1)
3.11.2.6	Gas Decay Tanks	5.5.10	YES	Although this Specification does not meet any Technical Specification selection criteria, it has been retained in accordance with the NRC letter from W. T. Russell to the industry ITS Chairpersons, dated October 25, 1993.
5.0	**DESIGN FEATURES**	**3.7.15** **4.0**	YES-2 YES	Application of Technical Specification selection criteria is not appropriate. However, specific portions of Design Features will be included in Technical Specifications as required by 10 CFR 50.36.
6.0	**ADMINISTRATIVE CONTROLS**	**5.0**	YES	Application of Technical Specification selection criteria is not appropriate. However, specific portions of Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.

[1] The Applicable Safety Analysis section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

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APPENDIX A

JUSTIFICATION FOR SPECIFICAITON RELOCATION

Appendix A - Justification For Specification Relocation

3.3.3.10, Explosive Gas Monitoring Instrumentation

DISCUSSION:

CTS 3.3.3.10 provides the requirements for the explosive gas monitoring instrumentation. This Specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the gaseous waste processing system is adequately monitored to ensure that the concentration is maintained below the flammability limit.

COMPARISON TO SCREENING CRITERIA:

1. Explosive gas monitoring instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. Explosive gas monitoring instrumentation is not used to indicate the status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. In addition, excessive system oxygen is not an indication of a DBA or transient.

3. Explosive gas monitoring instrumentation is not part of a primary success path in the mitigation of a DBA or transient. In addition, excessive oxygen discharge is not part of a primary success path in mitigating a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, the loss of the explosive gas monitoring instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. TVA has reviewed this evaluation, considers it applicable to Sequoyah Nuclear Plant (SQN) Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, Explosive Gas Monitoring Instrumentation LCO and Surveillances may be relocated to other plant controlled documents outside Technical Specifications.

Appendix A - Justification For Specification Relocation

3.6.2.2, Lower Containment Vent Coolers

DISCUSSION:

CTS 3.6.2.2 provides requirements on the Lower Containment Vent Coolers. The Lower Containment Vent Coolers are designed to maintain an acceptable temperature within the lower containment compartments for the protection of equipment and controls during normal reactor operation and normal shutdown.

COMPARISON TO SCREENING CRITERIA:

1. The Lower Containment Vent Coolers are not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The Lower Containment Vent Coolers are not a process variable, design feature, or operating restriction that is in 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 Lower Containment Vent Coolers are not a structure, system, or component that is part of the 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 Lower Containment Vent Coolers were found to be non-significant risk contributor to core damage frequency and offsite releases. Tennessee Valley Authority (TVA) has performed a plant-specific analysis to ensure that the Lower Containment Vent Coolers do not contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to be important to public health and safety.

CONCLUSION:

Since the screening criteria have not been satisfied, the Lower Containment Vent Coolers may be relocated to other plant controlled documents outside Technical Specifications.