Summary of Changes
ITS Split Report

Change Description	Affected Pages
The changes described in the KPS response to question RPG-010 have been made. This change clarifies adds the Containment Closure requirements to the ITS.	Pages 23 and 46 (information on page deleted due to change)
A typographical error has been corrected. The CTS reference should be 3.14.b, not 3.1.4.b.	Page 26

ATTACHMENT 1

VOLUME 1

KEWAUNEE POWER STATION IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

APPLICATION OF SELECTION CRITERIA TO THE KEWAUNEE POWER STATION TECHNICAL SPECIFICATIONS

Revision 1

APPLICATION OF SELECTION CRITERIA TO THE
KEWAUNEE POWER STATION
TECHNICAL SPECIFICATIONS

APPLICATION OF SELECTION CRITERIA
TO THE KEWAUNEE POWER STATION
TECHNICAL SPECIFICATIONS

CONTENTS Page

1. INTRODUCTION 1

2. SELECTION CRITERIA 2

3. PRA INSIGHTS 6

4. RESULTS OF APPLICATION OF SELECTION CRITERIA 9

5. REFERENCES 10

ATTACHMENT

1. SUMMARY DISPOSITION MATRIX FOR KPS

APPENDIX

A. JUSTIFICATION FOR SPECIFICATION RELOCATION

APPLICATION OF SELECTION CRITERIA TO THE KEWAUNEE POWER STATION 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 Kewaunee Power Station (KPS). Dominion Energy Kewaunee (hereinafter DEK) 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, Murley/Newton letter dated May 9, 1988 and as revised in NUREG-1431, Revision 3.0 "Standard Technical Specifications, Westinghouse Plants" (Reference 2) and applied the criteria to each of the current KPS 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 the KPS.

APPLICATION OF SELECTION CRITERIA
TO THE KEWAUNEE POWER STATION
TECHNICAL SPECIFICATIONS

2. SELECTION CRITERIA

DEK 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 DEK, 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 Updated Safety Analysis Report (USAR), for which a structure, system, or component must meet specified functional goals. These analyses are contained in Chapter 14 of the USAR (or equivalent chapters) and are identified as Condition II, III, or IV events

Attachment 1, Volume 1, Rev. 1, Page 5 of 55

APPLICATION OF SELECTION CRITERIA
TO THE KEWAUNEE POWER STATION
TECHNICAL SPECIFICATIONS

(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.

<u>Criterion 3:</u> 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:

<u>Discussion of Criterion 3:</u> 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,

APPLICATION OF SELECTION CRITERIA TO THE KEWAUNEE POWER STATION TECHNICAL SPECIFICATIONS

as presented in Chapters 14 of the plant's Updated 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).

<u>Criterion 4</u>: 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 4</u>: It is the Commission policy that licensees retain in their Technical Specifications LCOs, action statements and Surveillance Requirements for the following systems (as applicable), which operating experience and probabilistic risk assessment (PRA) have generally shown to be significant to public health and safety and any other structures, systems, or components that meet this criterion:

- Residual Heat Removal, 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.

**APPLICATION OF SELECTION CRITERIA
TO THE KEWAUNEE POWER STATION
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.

APPLICATION OF SELECTION CRITERIA
TO THE KEWAUNEE POWER STATION
TECHNICAL SPECIFICATIONS

3. PRA INSIGHTS

Introduction and Objectives

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. 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 DEK for the applicable KPS Specifications to be relocated. Where Reference 1 did not review a KPS Technical Specification against the criteria of Reference 3, DEK performed a review similar (but not identical) to that described below for Reference 1. The results of these reviews are presented in Appendix B.

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:

Attachment 1, Volume 1, Rev. 1, Page 9 of 55

APPLICATION OF SELECTION CRITERIA TO THE KEWAUNEE POWER STATION TECHNICAL SPECIFICATIONS

1. If the Technical Specification contained constraints of <u>prime importance</u> in limiting the likelihood or severity of the accident sequences that are <u>commonly found</u> to <u>dominate</u> 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 melt 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 melt, 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 melt frequency of 5×10^{-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 1×10^{-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 melt 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 in limiting the likelihood or severity of accident sequences that are commonly found to dominate risk.

APPLICATION OF SELECTION CRITERIA
TO THE KEWAUNEE POWER STATION
TECHNICAL SPECIFICATIONS

4. RESULTS OF APPLICATION OF SELECTION CRITERIA

The selection criteria from Section 2 were applied to the KPS 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. In addition, Appendix B includes a summary of the evaluations performed for those KPS specific Technical Specifications not evaluated in WCAP-11618.

**APPLICATION OF SELECTION CRITERIA
TO THE KEWAUNEE POWER STATION
TECHNICAL SPECIFICATIONS**

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 3.1, April 2001.

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

ATTACHMENT 1

SUMMARY DISPOSITION MATRIX
FOR
KEWAUNEE POWER STATION

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
1.0	**DEFINITIONS**	**1.0, 3.2.4, 3.6.1, 3.6.2, 3.6.8, 3.6.10, 3.7.12, 5.0, 6.0**	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 section 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 – Reactor Core	2.1		
2.1.a	Reactor Core	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.
2.1.b	DNBR	2.1.1.1	YES	Same as above.
2.1.c	Peak Centerline Temperature	2.1.1.2	YES	Same as above.
2.1.d	Safety Limit Compliance	2.2.1	YES	Same as above.
2.2	Safety Limit – Reactor Coolant System Pressure			
2.2.a	Reactor Coolant System Pressure	2.1.2	YES	Same as above.
2.2.b	Safety Limit Violation	2.2.2	YES	Same as above.
2.3	Limiting Safety System Settings – Protective Instrumentation	Relocated	NO	See technical change discussion in the Discussion of Changes for ITS 3.3.1.

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 15 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3.0	**LIMITING CONDITION FOR OPERATION**	**3.0**		
3.0.a	Compliance	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 3.
3.0.b	Noncompliance	LCO 3.0.2	YES	Same as above.
3.0.c	Operational Modes	LCO 3.0.3	YES	Same as above.
3.1	**REACTOR COOLANT SYSTEM**			
3.1.a	Operational Components			
3.1.a.1	Reactor Coolant Pumps			
3.1.a.1.A	Reactor Coolant Pumps Operation during Boron Concentration Reductions	3.4.5, 3.4.6, 3.4.7, 3.4.8, 3.9.4, 3.9.5	YES-3	
3.1.a.1.B	Reactor Coolant Pumps Operation	3.4.4	YES-2	
3.1.a.1.C	Starting Reactor Coolant Pumps	3.4.7, 3.4.12	YES-3	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 16 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3.1.a.2	Decay Heat Removal Capability			
3.1.a.2.A	Decay Heat Removal Capability	3.4.6	YES-3	
3.1.a.2.B	Decay Heat Removal Capability	3.4.7, 3.4.8, 3.9.4, 3.9.5	YES-4	
3.1.a.3	Pressurizer Safety Valves	3.4.10	YES-3	
3.1.a.4	Pressure Isolation Valves	3.4.14	YES-2	
3.1.a.5	Pressurizer Power-Operated Relief Valves (PORV) and PORV Block Valves	3.4.11	YES-3	
3.1.a.6	Pressurizer Heaters	3.4.9	YES-3	
3.1.a.7	Reactor Coolant Vent System	Relocated	NO	See Appendix A, Page 1.
3.1.b	Heatup and Cooldown Limit Curves for Normal Operation			
3.1.b.1	Reactor Coolant Temperature, Pressure, and System	3.4.3, 3.4.12	YES-2	
3.1.b.2	Steam Generators	Relocated	NO	See Appendix A, Page 2.
3.1.b.3	Pressurizer	Relocated	NO	See Appendix A, Page 3.
3.1.b.4	Overpressure Protection Systems for Low Temperature Operation	3.4.12	YES-2	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 17 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3.1.c	Maximum Coolant Activity			
3.1.c.1	Maximum Coolant Activity	3.4.16	YES-2	
3.1.c.2	Specific Activity Corrective Action	3.4.16	YES-2	
3.1.c.3	Specific Activity Reporting Requirements	Deleted	NO	See technical change discussion in the Discussion of Changes for ITS 3.4.16.
3.1.d	RCS Operational Leakage			
3.1.d.1	RCS Operational Leakage Limits	3.4.13	YES-2	
3.1.d.2	RCS Operational Leakage Limits	3.4.13	YES-2	
3.1.d.3	RCS Operational Leakage Limits	3.4.13	YES-2	
3.1.d.4	RCS Operational Leakage Detection Systems	3.4.15	YES-1	
3.1.e	Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration	Relocated	NO	See Appendix A, Page 4.
3.1.f	Minimum Conditions for Criticality			
3.1.f.1	Pressure Temperature Limits	3.4.3	YES-2	
3.1.f.2	Pressurizer Water Level	3.4.9	YES-2	
3.1.f.3	Moderator Temperature Coefficient (MTC) Limits	3.1.3, 3.1.8	YES-2	
3.1.f.4	Moderator Temperature Coefficient (MTC) Limits	3.1.3	YES-2	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 18 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3.1.g	Steam Generator (SG) Tube Integrity	3.4.20	YES-2	
3.2	**Chemical and Volume Control System**			
3.2.a	Chemical and Volume Control System	Relocated	NO	See Appendix A, Page 5.
3.3	**Engineered Safety Features and Auxiliary Systems**			
3.3.a	Accumulators	3.5.1	YES-3	
3.3.b	Emergency Core Cooling System			
3.3.b.1	Emergency Core Cooling System	3.5.2	YES-3	
3.3.b.2	Emergency Core Cooling System	3.5.2	YES-3	
3.3.b.3	Refueling Water Storage Tank	3.5.4	YES-3	
3.3.b.4	Refueling Water Storage Tank	3.5.4	YES-3	
3.3.b.5	Emergency Core Cooling System	3.5.2	YES-3	
3.3.c	Containment Cooling System			
3.3.c.1	Containment Spray and Containment Fancoil Units	3.6.6	YES-3	
3.3.c.2	Spray Additive System	3.6.7	YES-3	
3.3.d	Component Cooling System	3.7.7	YES-3	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 19 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3.3.e	Service Water System	3.3.2, 3.7.8	YES-3	
3.4	**Steam and Power Conversion System**			
3.4.a	Main Steam Safety Valves (MSSVs)	3.7.1	YES-3	
3.4.b	Auxiliary Feedwater System	3.7.5	YES-3	
3.4.c	Condensate Storage Tank	3.7.6	YES-2, 3	
3.4.d	Secondary Activity Limits	3.7.17	YES-2	
3.5	**Instrumentation System**			
3.5.a	Setting Limits for Instrumentation per Table 3.5-1	3.3.2, 3.3.5, 3.3.6	YSE-3	
3.5.b	On line testing or instrument failure per Table 3.5-2 through 3.5-5	3.3.1, 3.3.2, 3.3.5, 3.3.6	YES-3	
3.5.c	On line testing or instrument failure per Table 3.5-2 through 3.5-5	3.3.1, 3.3.2, 3.3.5, 3.3.6	YES-3	
3.5.d	On line testing or instrument failure per Table 3.5-2 through 3.5-5	3.3.1, 3.3.2, 3.3.5, 3.3.6	YES-3	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 20 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3.5.e	Accident Monitoring Instrumentation Table 3.5-6	3.3.3	YES-3	See Appendix A, Page 14. Instrumentation that does not monitor Regulatory Guide 1.97 Type A or Category 1 variables have been relocated in accordance with the guidance provided in NUREG-1431, Revision 3.0.
3.6	**Containment System**			
3.6.a	Containment System Integrity	3.6.1, 3.6.2, 3.6.8, 3.6.9	YES-3	
3.6.b.1	Containment Isolation Valves	3.6.3, 3.7.2, 3.7.3	YES-3	
3.6.b.2	Containment Isolation Valves	3.6.3	YES-3	
3.6.b.3	Containment Isolation Valves	3.6.3, 3.7.2, 3.7.3	YES-3	
3.6.b.4	Containment Isolation Valves	3.6.3, 3.7.2, 3.7.3	YES-3	
3.6.c	Conditions for Containment Integrity			
3.6.c.1	Shield Building Ventilation System	3.6.10	YES-3	
3.6.c.2	Auxiliary Building Special Ventilation System	3.7.12	YES-3	
3.6.c.3	Performance Requirements	5.5	YES-3	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 21 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3.6.d	Containment Internal Pressure	3.6.4	YES-2	
3.6.e	Containment Ambient Temperature	3.6.5	YES-2	
3.7	**Auxiliary Electrical Systems**			
3.7.a.1	Reserve Auxiliary Transformer	3.8.1	YES-3	
3.7.a.2	External Source of Power to Emergency Buses 1-5 and 1-6	3.8.1	YES-3	
3.7.a.3	4160-V Buses 1-5 and 1-6	3.8.9	YES-3	
3.7.a.4	480-V Buses 1-52 and 1-62	3.8.9	YES-3	
3.7.a.5	480-V Buses 1-51 and 1-61	3.8.9	YES-3	
3.7.a.6	Station Batteries and DC Systems	3.8.4, 3.8.6, 3.8.9	YES-3	
3.7.a.7	Diesel Generators	3.8.1, 3.8.3	YES-3	
3.7.a.8	Transmission Lines	3.8.1	YES-1	
3.7.b.1	Condition for Inoperability	3.8.1	YES-3	
3.7.b.2	Condition for Inoperability	3.8.1	YES-3	
3.7.b.3	Condition for Inoperability	3.8.4, 3.8.6	YES-3	
3.7.b.4	Condition for Inoperability	3.8.1	YES-3	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 22 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3.7.b.5	Condition for Inoperability	3.8.1	YES-3	
3.7.b.6	Condition for Inoperability	3.8.9	YES-3	
3.7.b.7	Condition for Inoperability	3.8.1	YES-3	
3.7.c	Condition for Inoperability	3.8.1	YES-3	
3.8	**Refueling Operations**			
3.8.a.1	Containment Closure	3.9.6, 3.3.6	YES-3	
3.8.a.2	Radiation Levels Monitoring	Relocated	NO	See Appendix A, Page 7.
3.8.a.3	Neutron Monitoring	3.9.2, Relocated	YES-3	See technical change discussion in the Discussion of Changes for CTS 3.8.a.3.
3.8.a.4	Residual Heat Removal Pump Operability	3.9.4	YES-4	
3.8.a.5	Boron Concentration	3.9.1	YES-2	
3.8.a.6	Direct Communications	Relocated	NO	See Appendix A, Page 8.
3.8.a.7	Deleted			
3.8.a.8	Containment Ventilation and Purge System	3.9.6	YES-3	
3.8.a.9	Spent Fuel Pool Sweep System	Relocated	NO	See Appendix A, Page 9.
3.8.a.10	Minimum Water Level Above the Flange	3.9.6	YES-2	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 23 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3.8.a.11	Dead Load Test	Relocated	NO	See Appendix A, Page 10.
3.8.a.12	Licensed Operator (SRO) in charge of Refuel Operations	Deleted	NO	See technical change discussion in the Discussion of Changes for CTS 3.8.a.12.
3.8.b	Refueling Operations Conditions not met	3.3.6, 3.9.1, 3.9.2, 3.9.3, 3.9.4, 3.9.6	YES-2, 3	
3.10	**Control Rod and Power Distribution Limits**			
3.10.a	Shutdown Reactivity	3.1.1	YES-2	
3.10.b	Power Distribution Limits			
3.10.b.1.A	Heat Flux Hot Channel Factor	3.2.1	YES-2	
3.10.b.1.B	Nuclear Enthalpy Rise Hot Channel Factor	3.2.2	YES-2	
3.10.b.2	Nuclear Enthalpy Rise Hot Channel Factor Actions	3.2.2	YES-2	
3.10.b.3	Heat Flux Hot Channel Factor Actions	3.2.1	YES-2	
3.10.b.4	Heat Flux and Nuclear Enthalpy Rise Hot Channel Factors Actions	3.2.1, 3.2.2	YES-2	
3.10.b.5	Heat Flux Hot Channel Factor Actions	3.2.1	YES-2	
3.10.b.6	Heat Flux Hot Channel Factor Actions	3.2.1	YES-2	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 24 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3.10.b.7	Heat Flux Hot Channel Factor Actions	3.2.1	YES-2	
3.10.b.8	Axial Flux Difference	3.2.3	YES-2	
3.10.c	Quadrant Power Tilt Limits	3.2.4	YES-2	
3.10.d	Rod Insertion Limits			
3.10.d.1	Rod Insertion Limits	3.1.5	YES-2	
3.10.d.2	Rod Insertion Limits	3.1.6	YES-2	
3.10.d.3	Rod Insertion Limits	3.1.5, 3.1.6, 3.1.8	YES-2	
3.10.e	Rod Misalignment Limitations	3.1.4, 3.1.8	YES-2	
3.10.f	Inoperable Rod Position Indicator Channels	3.1.4, 3.1.7	YES-2	
3.10.g	Inoperable Rod Limitations	3.1.4	YES-2	
3.10.h	Rod Drop Time	3.1.4	YES-2	
3.10.i	Rod Position Deviation Monitor	Deleted	NO	See technical change discussion in the Discussion of Changes for ITS 3.1.4.
3.10.j	Quadrant Power Tilt Monitor	Deleted	NO	See technical change discussion in the Discussion of Changes for ITS 3.2.4.
3.10.k	Core Average Temperature	3.4.1	YES-2	
3.10.l	Reactor Coolant System Pressure	3.4.1	YES-2	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 25 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3.10.m	Reactor Coolant Flow	3.4.1	YES-2	
3.10.n	DNBR Parameters	3.4.1	YES-2	
3.11	**Core Surveillance Instrumentation**			
3.11.a	Core Surveillance Instrumentation	Relocated	NO	See Appendix A, Page 11.
3.11.b	Core Surveillance Instrumentation	Relocated	NO	See Appendix A, Page 11.
3.11.c	Core Surveillance Instrumentation	3.3.3	YES-3	
3.11.d	Core Surveillance Instrumentation	Relocated	NO	See Appendix A, Page 11.
3.12	**Control Room Post-Accident Recirculation System**			
3.12.a	Control Room Post-Accident Recirculation System	3.3.7, 3.7.10	YES-3	
3.12.b	Control Room Post-Accident Recirculation System	3.3.7, 3.7.10	YES-3	
3.12.c	Control Room Post-Accident Recirculation System	5.5	YES-3	
3.14	**Shock Suppressors (Snubbers)**			
3.14.a	Shock Suppressors (Snubbers)	Relocated	NO	See technical change discussion in the Discussion of Changes for CTS 3.14.
3.14.b	Shock Suppressors (Snubbers) Actions	SR 3.0.8	YES	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 26 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
4.0	**Surveillance Requirements**			
4.0.a	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 3.
4.0.b	Time of Performance	SR 3.0.2	YES	Same as above.
4.0.c	Noncompliance	SR 3.0.3	YES	Same as above.
4.0.d	Entry into Operational Modes	SR 3.0.4	YES	Same as above.
4.1	**Operational Safety Review**			
4.1.a	Calibration, testing and checking of instrumentation	3.1.7, 3.3.1, 3.3.2, 3.3.3, 3.3.5, 3.3.6, 3.3.7, 3.4.15	YES-2, 3	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 27 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
4.1.b	Equipment and Sampling Tests	3.1.1, 3.1.4, 3.3.1, 3.4.9, 3.4.11, 3.4.13, 3.4.16, 3.5.1, 3.5.4, 3.8.3	YES-2, 3	
4.2	**ASME Code Class Inservice Inspection and Testing**			
4.2.a.1	ASME Code Class Inservice Inspection and Testing	Relocated	NO	See technical change discussion in the Discussion of Change for ITS 5.5.6.
4.2.a.2	ASME Code Class Inservice Inspection and Testing	5.5	YES	Retained as a program in the Administrative Controls Section.
4.2.a.3	Surveillance Testing of Pressure Isolation Valves	3.4.14	YES-2	
4.4	**Containment Tests**			
4.4.a	Integrated Leak Rate Tests (Type A)	3.6.1	YES-3	
4.4.b	Local Leak Rate Tests (Type B and C)	3.6.1, 3.6.2, 3.6.3	YES-3	
4.4.c.1	Shield Building Ventilation System	3.6.10, 3.7.12, 5.5	YES-3	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 28 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
4.4.c.2	Shield Building Ventilation System Filter Testing	5.5	YES-3	
4.4.c.3	Shield Building Ventilation System Testing	5.5	YES-3	
4.4.c.4	Shield Building Ventilation System Testing	3.6.8	YES-3	
4.4.d	Auxiliary Building Special Ventilation System	3.7.12, 5.5	YES-3	
4.4.e	Containment Vacuum Breaker System	3.6.1, 3.6.9	YES-3	
4.4.f	Containment Isolation Device Position Verification	3.6.3	YES-3	
4.5	**Emergency Core Cooling System and Containment Air Cooling System Tests**			
4.5.a.1	System Tests – Safety Injection Systems	3.5.2	YES-3	
4.5.a.2	System Tests – Containment Vessel Internal Spray System	3.6.6	YES-3	
4.5.a.3	System Tests – Containment Fancoil Units	3.6.6	YES-3	
4.5.b.1	Component Tests – Pumps	3.5.2, 3.6.6	YES-3	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 29 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
4.5.b.2	Component Tests – Valves	3.4.14, 3.5.1, 3.5.2, 3.6.7	YES-3	
4.6	**Periodic Testing of Emergency Power System**			
4.6.a	Diesel Generators	3.3.5, 3.8.1	YES-3	
4.6.b	Station Batteries	3.8.6	YES-3	
4.7	**Main Steam Isolation Valves**	3.7.2	YES-3	
4.8	**Auxiliary Feedwater System**	3.7.5	YES-3	
4.9	**Reactor Anomalies**	3.1.2	YES-2	
4.12	**Spent Fuel Pool Sweep System**	Relocated	NO	See Appendix A, Page 9.
4.13	**Radioactive Materials Sources**	Relocated	NO	See Appendix A, Page 12.
4.14	**Testing and Surveillance of Shock Suppressors (Snubbers)**	Relocated	NO	See technical change discussion in the Discussion of Changes for CTS 3.14.
4.16	**Reactor Coolant Vent System Tests**	Relocated	NO	See Appendix A, Page 1.
4.17	**Control Room Postaccident Recirculation System**	3.7.10, 5.5	YES-3	
4.18	**RCS Operational LEAKAGE**	3.4.13	YES-2	
4.19	**Steam Generator (SG) Tube Integrity**	3.4.20	YES-2	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 30 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
Table TS 4.1-1	**Minimum Frequencies for Checks, Calibrations and Test of Instrument Channels**			Table TS 4.1-1 provides the calibration, testing, and checking requirements for instrument channels and testing of logic channels. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the requirements of Table TS 4.1-1 will be retained in Technical Specifications, as modified consistent with NUREG-1431, Revision 3.
Table TS 4.1-1 Channel Description 1	Nuclear Power Range	3.3.1	YES	
Table TS 4.1-1 Channel Description 2	Nuclear Intermediate Range	3.3.1	YES	
Table TS 4.1-1 Channel Description 3	Nuclear Source Range	3.3.1, 3.9.2	YES	
Table TS 4.1-1 Channel Description 4	Reactor Coolant Temperature	3.3.1	YES	
Table TS 4.1-1 Channel Description 5	Reactor Coolant Flow	3.3.1	YES	
Table TS 4.1-1 Channel Description 6	Pressurizer Water Level	3.3.1	YES	
Table TS 4.1-1 Channel Description 7	Pressurizer Pressure	3.3.1, 3.3.2	YES	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 31 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
Table TS 4.1-1 Channel Description 8.a	4-kV Voltage and Frequency	3.3.1	YES	
Table TS 4.1-1 Channel Description 8.b	4-kV Voltage (Loss of Voltage)	3.3.5	YES	
Table TS 4.1-1 Channel Description 8.c	4-kV Voltage (Degraded Grid)	3.3.5	YES	
Table TS 4.1-1 Channel Description 9	Analog Rod Position	3.1.4, 3.1.7	YES	
Table TS 4.1-1 Channel Description 10	Rod Position Bank Counters	3.1.4, 3.1.7	YES	
Table TS 4.1-1 Channel Description 11.a	Steam Generator Low Level	3.3.1, 3.3.2	YES	
Table TS 4.1-1 Channel Description 11.b	Steam Generator High Level	3.3.2	YES	
Table TS 4.1-1 Channel Description 12	Steam Generator Flow Mismatch	3.3.1	YES	
Table TS 4.1-1 Channel Description 14	Residual Heat Removal Pump Flow	Relocated	NO	See technical change discussion in the Discussion of Change for ITS 3.3.2.

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 32 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
Table TS 4.1-1 Channel Description 16	Refueling Water Storage Tank Level	3.3.3	YES	
Table TS 4.1-1 Channel Description 18.a	Containment Pressure (SIS signal)	3.3.2	YES	
Table TS 4.1-1 Channel Description 18.b	Containment Pressure (Steamline Isolation)	3.3.2	YES	
Table TS 4.1-1 Channel Description 18.c	Containment Pressure (Containment Spray Act)	3.3.2	YES	
Table TS 4.1-1 Channel Description 18.d	Annulus Pressure (Vacuum Breaker)	3.6.9	YES	
Table TS 4.1-1 Channel Description 19	Radiation Monitoring System	3.3.2, 3.3.6, 3.3.7, 3.4.15, Relocated	YES	See technical change discussion in the Discussion of Change for ITS 3.3.2.
Table TS 4.1-1 Channel Description 21	Containment Sump Level	3.4.15	YES	
Table TS 4.1-1 Channel Description 22	Accumulator Level and Pressure	3.5.1	YES	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 33 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
Table TS 4.1-1 Channel Description 23	Steam Generator Pressure	3.3.2	YES	
Table TS 4.1-1 Channel Description 24	Turbine First Stage Pressure	3.3.1	YES	
Table TS 4.1-1 Channel Description 26	Protective System Logic Channel Testing	3.3.1, 3.3.2, 3.8.1	YES	
Table TS 4.1-1 Channel Description 29	Seismic Monitoring	Relocated	NO	See Appendix A, Page 13.
Table TS 4.1-1 Channel Description 30	Fore Bay Water Level	3.7.8	YES	
Table TS 4.1-1 Channel Description 31	AFW Flow Rate	Relocated	NO	See Appendix A, Page 14.
Table TS 4.1-1 Channel Description 32	PORV Position Indication	Relocated	NO	See Appendix A, Page 14.
Table TS 4.1-1 Channel Description 32.a	PORV Position Indication – Back-up (Temperature)	Relocated	NO	See Appendix A, Page 14.
Table TS 4.1-1 Channel Description 33	PORV Block Valve Position Indicator	Relocated	NO	See Appendix A, Page 14.

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 34 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
Table TS 4.1-1 Channel Description 34	Safety Valve Position Indicator (Acoustic)	Relocated	NO	See Appendix A, Page 14.
Table TS 4.1-1 Channel Description 34.a	Safety Valve Position Indicator – Back-up (Temperature)	Relocated	NO	See Appendix A, Page 14.
Table TS 4.1-1 Channel Description 35	FW Pump Trip (AFW Initiation)	3.3.2	YES	
Table TS 4.1-1 Channel Description 36	Reactor Coolant System Subcooling Monitor	3.3.3	YES	
Table TS 4.1-1 Channel Description 37	Containment Pressure (Wide Range)	3.3.3	YES	
Table TS 4.1-1 Channel Description 39	Containment Water Level (Wide Range)	3.3.3	YES	
Table TS 4.1-1 Channel Description 40	Reactor Vessel Level Indication	3.3.3	YES	
Table TS 4.1-1 Channel Description 41	Core Exit Thermocouples	3.3.3	YES	
Table TS 4.1-1 Channel Description 42	Steam Generator Level (Wide Range)	3.3.3	YES	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 35 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
Table TS 4.1-1 Channel Description 43	AFW Pump Low Discharge Pressure Trip	Relocated	NO	See Technical change discussion in the Discussion of Change for ITS 3.7.5.
Table TS 4.1-1 Channel Description 44	Axial Flux Difference (AFD)	3.2.3	YES	
Table TS 4.1-1 Channel Description 45	Service Water Turbine Header Isolation Logic Trip (SW 4 A/B)	3.3.2	YES	
Table TS 4.1-1 Channel Description 46	AFW Pump Low Suction Pressure Trip	Relocated	NO	See technical change discussion in the Discussion of Change for ITS 3.7.5.
Table TS 4.1-2	**Minimum Frequencies for Sampling Tests**			Table TS 4.1-2 provides the sampling test requirements for one or more Specifications. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the requirements of Table TS 4.1-2 will be retained in Technical Specifications, as modified consistent with NUREG-1431, Revision 3.
Table TS 4.1-2 Sampling Test 1	Reactor Coolant Samples	3.1.1, 3.4.16, Relocated	YES	See technical change discussion in the Discussion of Change for CTS 3.1.e.
Table TS 4.1-2 Sampling Test 2	Reactor Coolant Boron	3.1.1, 3.9.1	YES	
Table TS 4.1-2 Sampling Test 3	Refueling Water Storage Tank Water Sample	3.5.4	YES	
Table TS 4.1-2 Sampling Test 5	Accumulator	3.5.1	YES	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 36 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
Table TS 4.1-2 Sampling Test 6	Spent Fuel Pool	3.7.15	YES	
Table TS 4.1-2 Sampling Test 7	Secondary Coolant	3.7.17	YES	
Table TS 4.1-3	**Minimum Frequencies for Equipment Tests**			Table TS 4.1-3 provides the equipment test requirements for one or more Specifications. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the requirements of Table TS 4.1-3 will be retained in Technical Specifications, as modified consistent with NUREG-1431, Revision 3.
Table TS 4.1-3 Equipment Test 1	Control Rods	3.1.4	YES	
Table TS 4.1-3 Equipment Test 1.a	Reactor Trip Breakers	3.3.1	YES	
Table TS 4.1-3 Equipment Test 1.b	Reactor Coolant Pump Breakers – Open-Reactor Trip	3.3.1	YES	
Table TS 4.1-3 Equipment Test 1.c	Manual Reactor Trip	3.3.1	YES	
Table TS 4.1-3 Equipment Test 4	Containment Isolation Trip	3.6.3	YES	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 37 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
Table TS 4.1-3 Equipment Test 5	Refueling System Interlocks	Relocated	NO	See Appendix A, Page 10.
Table TS 4.1-3 Equipment Test 8	RCS Leak Detection	3.4.15	YES	
Table TS 4.1-3 Equipment Test 9	Diesel Fuel Supply	3.8.1, 3.8.3	YES	
Table TS 4.1-3 Equipment Test 11	Fuel Assemblies	Deleted	NO	See technical change discussion in the Discussion of Change for ITS 4.0.
Table TS 4.1-3 Equipment Test 12	Guard Pipes	Relocated	No	See technical change discussion in the Discussion of Change for ITS 3.6.1.
Table TS 4.1-3 Equipment Test 13	Pressurizer PORVs	3.4.11	YES	
Table TS 4.1-3 Equipment Test 14	Pressurizer PORV Block Valves	3.4.11	YES	
Table TS 4.1-3 Equipment Test 15	Pressurizer Heaters	3.4.9	YES	
Table TS 4.1-3 Equipment Test 16	Containment Purge and Vent Isolation Valves	3.6.3	YES	

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 38 of 55

SUMMARY DISPOSITION MATRIX FOR KEWAUNEE POWER STATION

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
5.0	**Design Features**	3.7.15, 3.7.16, 4.1, 4.2, 4.3	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**	3.6.3, 5.1, 5.2, 5.3, 5.4, 5.5, 5.6, 5.7, Relocated	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. See technical change discussion in the Discussion of Change for CTS 6.0 for relocated Specifications

(a) The Applicable Safety Analyses section of the Bases for the individual Technical Specifications describes the reason specific Technical Specification selection criteria are met.

Attachment 1, Volume 1, Rev. 1, Page 39 of 55

APPENDIX A

JUSTIFICATION FOR SPECIFICATION RELOCATION

Appendix A - Justification For Specification Relocation

3.1.a.7: Reactor Coolant Vent System
4.16: Reactor Coolant Vent System Tests

DISCUSSION:

CTS 3.1.a.7 provides requirements for the reactor coolant vent system. CTS 4.16 provides the testing requirements for the reactor coolant vent system.

COMPARISON TO SCREENING CRITERIA:

1. The reactor coolant vent system is not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The reactor coolant vent system is 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 reactor coolant vent system is 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 reactor coolant vent system is not a structure, system or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the screening criteria have not been satisfied, the reactor coolant vent system Specifications may be relocated to other plant controlled documents outside Technical Specifications.

Appendix A - Justification For Specification Relocation

3.1.b.2: Steam Generators

DISCUSSION:

CTS 3.1.b.2 states that the secondary side of the steam generator must not be pressurized > 200 psig if the temperature of the steam generator is < 70ºF.

COMPARISON TO SCREENING CRITERIA:

1. The steam generator pressure/temperature limit is not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The steam generator pressure/temperature limit is 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 steam generator pressure/temperature limit is 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 steam generator pressure/temperature limit is not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the screening criteria have not been satisfied, the Steam Generator pressure/temperature limits Specification may be relocated to other plant controlled documents outside Technical Specifications.

Appendix A - Justification For Specification Relocation

3.1.b.3: Pressurizer

DISCUSSION:

CTS 3.1.b.3 states that the pressurizer cooldown and heatup rates shall not exceed 200ºF/hr and 100ºF/hr, respectively. It also states that the spray shall not be used if the temperature difference between the pressurizer and the spray fluid is > 320ºF.

COMPARISON TO SCREENING CRITERIA:

1. The pressurizer pressure/temperature limits 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.

2. The pressurizer pressure/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 pressure/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 pressure/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.

CONCLUSION:

Since the screening criteria have not been satisfied, the Pressurizer Specification may be relocated to other plant controlled documents outside Technical Specifications.

Attachment 1, Volume 1, Rev. 1, Page 43 of 55

Appendix A - Justification For Specification Relocation

3.1.e: Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration
CTS Table TS 4.1-2 Sampling Test 1.d: Reactor Coolant Samples

DISCUSSION:

CTS 3.1.e provides limits on the oxygen, chloride and fluoride content in the RCS. CTS Table TS 4.1-2 Sampling Test 1.d provides the testing requirements for the oxygen, chloride and fluoride content in the RCS.

COMPARISON TO SCREENING CRITERIA:

1. The RCS chemistry limits 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.

2. The RCS chemistry 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 RCS chemistry limits 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 RCS chemistry 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.

CONCLUSION:

Since the screening criteria have not been satisfied, the RCS Chemistry Specification may be relocated to other plant controlled documents outside Technical Specifications.

Attachment 1, Volume 1, Rev. 1, Page 44 of 55

Appendix A - Justification For Specification Relocation

3.2.a: Chemical and Volume Control System

DISCUSSION:

CTS 3.2.a provides requirements on boric acid flow paths to the reactor core although the specification is labeled "Chemical and Volume Control System." The purpose for a boric acid flow path is for control of the chemical neutron absorber (boron) concentration in the Reactor Coolant System (RCS) and to help maintain the SHUTDOWN MARGIN.

COMPARISON TO SCREENING CRITERIA:

1. The Chemical and Volume Control System is not used to detect, and indicate in the control room, a significant degradation of the reactor coolant pressure boundary.

2. The Chemical and Volume Control System is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The Chemical and Volume Control System is 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 loss of the Chemical and Volume Control System (i.e., boric acid flow path to the core – operating and shutdown) was found to be a non-significant risk contributor to core damage frequency and offsite releases.

CONCLUSION:

Since the screening criteria have not been satisfied, the Chemical and Volume Control System Specification may be relocated to other plant controlled documents outside Technical Specifications.

Appendix A - Justification For Specification Relocation

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Appendix A - Justification For Specification Relocation

3.8.a.2: Area Radiation Monitors

DISCUSSION:

CTS 3.8.a.2 provides the requirements for continuously monitoring radiation levels in the fuel handling area, the containment, and the spent fuel pool storage pool.

COMPARISON TO SCREENING CRITERIA:

1. The area radiation monitoring is not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. The area radiation monitoring is 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 area radiation monitoring is 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 area radiation monitoring is not a structure, system or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the screening criteria have not been satisfied, the Area Radiation Monitors may be relocated to other plant controlled documents outside Technical Specifications.

Appendix A - Justification For Specification Relocation

3.8.a.6: Direct Communications

DISCUSSION:

CTS 3.8.a.6 states that direct communication between the control room and the operating floor of the containment shall be available whenever changes in core geometry are taking place. This ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during changes in core geometry.

COMPARISON TO SCREENING CRITERIA:

1. Communications 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.

2. Communications are not a process variable, design feature, or operating restriction that is an initial condition of a Design Basis Analysis (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Communications are not a structure, system, or component that is part of a 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. Communications are not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the screening criteria have not been satisfied, the Direct Communications Specification may be relocated to other plant controlled documents outside Technical Specifications.

Appendix A - Justification For Specification Relocation

3.8.a.9: Spent Fuel Pool Sweep System
4.12: Spent Fuel Pool Sweep System

DISCUSSION:

CTS 3.8.a.9 and 4.12 provide requirements on the Spent Fuel Pool Sweep System. CTS Table TS 4.1-1 Channel Description 19 (Radiation Monitors R13 and R14 only) provides the testing requirements for the Auxiliary Building Vent Monitors used to initiate closure of the ventilation dampers for the Spent Fuel Pool Sweep System. The purpose of the Spent Fuel Pool Sweep System is to filter radioactive particulates from the area of the fuel pool. The purpose of the Auxiliary Building vent monitors is to monitor the Auxiliary Building vent flowpath on a continuous basis.

COMPARISON TO SCREENING CRITERIA:

1. The Spent Fuel Pool Sweep System and the Auxiliary Building Vent Monitors 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 Spent Fuel Pool Sweep System and the Auxiliary Building Vent Monitors 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 Spent Fuel Pool Sweep System and the Auxiliary Building Vent Monitors 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 Spent Fuel Pool Sweep System and the Auxiliary Building Vent Monitors were found to be non-significant risk contributor to core damage frequency and offsite releases.

CONCLUSION:

Since the screening criteria have not been satisfied, the Spent Fuel Pool Sweep System and the Auxiliary Building Vent Monitors Specifications may be relocated to other plant controlled documents outside Technical Specifications.

Attachment 1, Volume 1, Rev. 1, Page 49 of 55

Appendix A - Justification For Specification Relocation

3.8.a.11: Dead Load Test
Table TS 4.1-3, Equipment Test 5: Refueling System Interlocks

DISCUSSION:

CTS 3.8.a.11 states that a dead-load test shall be successfully performed on both the fuel handling and manipulator cranes before fuel movement begins. The load assumed by the cranes for the test must be equal to or greater than the maximum load to be assumed by the cranes during the REFUELING OPERATIONS. CTS 3.8.a.11 also requires a thorough visual inspection of the cranes shall be made after the dead-load test and prior to fuel handling. This Specification ensures the lifting device on the Manipulator Crane has adequate capacity to lift the weight of a fuel assembly and a Rod Control Cluster Assembly, and that an automatic load limiting device is available to prevent damage to the fuel assembly during fuel movement. This Specification also ensures the auxiliary hoist on the Manipulator Crane has adequate capacity for latching and unlatching control rod drive shafts. CTS Table TS 4.1-3, Equipment Test 5 requires the Refueling System Interlocks to be tested each Refueling Outage. This test ensures the other manipulator crane interlocks (e.g., ensuring only one component can be moved at one time) are OPERABLE.

COMPARISON TO SCREENING CRITERIA:

1. The Fuel Handling and Manipulator Cranes 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.

2. The Fuel Handling and Manipulator Cranes are not a process variable, design feature, or operating restriction that is an initial condition of a Design Basis Analysis (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The Fuel Handling and Manipulator Cranes are not a structure, system, or component that is part of the primary success path and which functions to actuate or 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 Fuel Handling and Manipulator Cranes are not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the screening criteria have not been satisfied, the Dead Load Test Specification may be relocated to other plant controlled documents outside Technical Specifications.

Appendix A - Justification For Specification Relocation

3.11.a: Core Surveillance Instrumentation
3.11.b: Core Surveillance Instrumentation
3.11.d: Core Surveillance Instrumentation

DISCUSSION:

CTS 3.11, 3.11.a, 3.11.b, and 3.11.d ensure the OPERABILITY of movable incore detector instrumentation when required to monitor the flux distribution within the core. The instrumentation is used for periodic Surveillance of the reactor core power distribution, and calibration of the excore neutron flux detectors, but is not assumed in any design basis accident (DBA) analysis and does not mitigate an accident.

COMPARISON TO SCREENING CRITERIA:

1. The movable incore detectors 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.

2. The movable incore detectors 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 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 DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The movable incore detectors are not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the screening criteria have not been satisfied, the Core Surveillance Instrumentation Specification may be relocated to other plant controlled documents outside Technical Specifications.

Appendix A - Justification For Specification Relocation

4.13: Radioactive Materials Sources

DISCUSSION:

CTS 4.13 provides the testing requirements for possession, leak test, and record requirements for radioactive material sources required for operation of the facility. The limitations of sealed source contamination are intended to ensure that the radioactive material sources are available to the facility and are free from leakage. These Surveillance Requirements bear no relation to the conditions or limitations that are necessary to ensure safe reactor operation.

COMPARISON TO SCREENING CRITERIA:

1. Radioactive Materials Sources is not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. Radioactive Materials Sources is 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. Radioactive Materials Sources is 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. Radioactive Materials Sources is not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the screening criteria have not been satisfied, the Radioactive Materials Sources Specification may be relocated to other plant controlled documents outside Technical Specifications.

Appendix A - Justification For Specification Relocation

Table TS 4.1-1, Channel Description 29: Seismic Instrumentation

DISCUSSION:

CTS Table TS 4.1-1 Channel Description 29 provides requirements for seismic instrumentation. In the event of an earthquake, seismic instrumentation is required to permit comparison of the measured response to that used in the design basis of the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR 100. Since this is determined after the event has occurred, it has no bearing on the mitigation of any design basis accident (DBA). The ITS does not include this Specification.

COMPARISON TO SCREENING CRITERIA:

1. Seismic instrumentation is not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. Seismic instrumentation is 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. Seismic instrumentation is not a structure, system, or component that is part of a 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. Seismic instrumentation is not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

CONCLUSION:

Since the screening criteria have not been satisfied, the seismic instrumentation may be relocated out of the Technical Specifications.

Attachment 1, Volume 1, Rev. 1, Page 53 of 55

Appendix A - Justification For Specification Relocation

Table TS 3.5-6: Accident Monitoring Instrumentation Operating Conditions for Indication
Table TS 4.1-1 Channel Description 31: AFW Flow Rate
Table TS 4.1-1 Channel Description 32: PORV Position Indication
Table TS 4.1-1 Channel Description 32.a: PORV Position Indication – Back-up (Temperature)
Table TS 4.1-1 Channel Description 33: PORV Block Valve Position Indicator
Table TS 4.1-1 Channel Descriptions 34: Pressurizer Safety Valve Position Indicator (Acoustic)
Table TS 4.1-1 Channel Descriptions 34.a: Pressurizer Safety Valve Position Indicator – Back-up (Temperature)

DISCUSSION:

CTS Tables TS 3.5-6 and TS 4.1-1 provide requirements for Post-Accident Monitoring Instrumentation channels. Each individual post accident monitoring parameter has a specific purpose; however, the general purpose for all accident monitoring instrumentation is to ensure sufficient information is available following an accident to allow an operator to verify the response of automatic safety systems, and to take preplanned manual actions to accomplish a safe shutdown of the plant.

The NRC position on application of the screening criteria to post-accident monitoring instrumentation is documented in a letter dated May 9, 1988 from T.E. Murley (NRC) to W.S. Wilgus (B&W Owners Group). The screening criteria are now incorporated into 10 CFR 50.36(c)(2)(ii). The NRC position taken was that the post-accident monitoring instrumentation table list should contain, on a plant specific basis, all Regulatory Guide 1.97 Type A plant instruments specified in the plant's Safety Evaluation Report (SER) on Regulatory Guide 1.97, and all Regulatory Guide 1.97 Category 1 plant instruments. Accordingly, this position has been applied to KPS Regulatory Guide 1.97 plant instruments. Those plant instruments meeting these criteria have remained in Technical Specifications. The instruments not meeting these criteria will be relocated from the Technical Specifications to the Technical Requirements Manual (TRM).

A review of the KPS USAR and the NRC Regulatory Guide 1.97 Safety Evaluation for KPS shows that the following CTS Table TS 3.5-6 and TS 4.1-1 Instruments do not meet Category 1 or Type A requirements.

Functional Unit 1/Channel Description 31, AFW Flow Rate
Functional Unit 3/Channel Descriptions 32 and 32.a, PORV Position
Functional Unit 4/Channel Description 33, PORV Block Valve Position
Functional Unit 5/Channel Descriptions 34 and 34.a, Pressurizer Safety Valve Position

COMPARISON TO SCREENING CRITERIA:

1. These instruments 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.

2. The monitored parameters are not process variables, design features, or operating restrictions that are initial conditions 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. These instruments are not a structure, system, or component that is part of a 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. These instruments are not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Appendix A - Justification For Specification Relocation

CONCLUSION:

Since the screening criteria have not been satisfied, the above listed instruments may be relocated out of the Technical Specifications.