VOLUME 1

CNP UNITS 1 AND 2
IMPROVED TECHNICAL
SPECIFICATIONS CONVERSION

APPLICATION OF SELECTION CRITERIA
TO THE
DONALD C. COOK NUCLEAR PLANT
UNITS 1 AND 2
TECHNICAL SPECIFICATIONS

Revision 0

APPLICATION OF SELECTION CRITERIA
TO THE DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2
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APPLICATION OF SELECTION CRITERIA TO THE DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2 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 Donald C. Cook Nuclear Plant (CNP) Units 1 and 2. Indiana Michigan Power Company (I&M) 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 2 "Standard Technical Specifications, Westinghouse Plants" (Reference 2) and applied the criteria to each of the current CNP Units 1 and 2 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 CNP Units 1 and 2.

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2. SELECTION CRITERIA

I&M 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 I&M, 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 (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).

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2. SELECTION CRITERIA (continued)

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

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2. SELECTION CRITERIA (continued)

- 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 PSA'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 PSA 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 PSA or risk survey and any available literature on risk insights and PSAs. 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 PSAs 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 PSA or risk survey and any available literature on risk insights and PSAs 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 I&M for the applicable CNP Units 1 and 2 Specifications to be relocated. Where Reference 1 did not review a CNP Units 1 and 2 Technical Specification against the criteria of Reference 3, I&M 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:

   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 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.

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3. PRA INSIGHTS (continued)

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

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

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

Systems identified in Tables 3 and 4 of Section 4.0 of WCAP-11618 that contributed significantly to risk as defined in Paragraph d above were listed in Tables 3A, 3B, 4A and 4B of Section 4.0. These identified systems as well as sequences and the risk dominant initiating events from Tables 3 and 4 which were involved in typical dominant core 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 <u>prime importance</u> to limiting the likelihood or severity of sequences that are <u>commonly found</u> to <u>dominate</u> risk.

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4. RESULTS OF APPLICATION OF SELECTION CRITERIA

The selection criteria from Section 2 were applied to the CNP Units 1 and 2 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 CNP Units 1 and 2 specific Technical Specifications not evaluated in WCAP-11618.

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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 2, April 2001.

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

ATTACHMENT 1

SUMMARY DISPOSITION MATRIX
FOR
CNP UNITS 1 AND 2

SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[(a)]
1.0	**DEFINITIONS**	**1.1**	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	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.
2.1.2	Reactor Coolant System Pressure	2.1.2	YES	Same as above.
2.2	Limiting Safety System Settings	3.3.1		
2.2:1	Reactor Trip System Instrumentation Setpoints	3.3.1	YES-3	The RTS LSSS have been included as part of the RTS 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.
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 2.
3.0.2	Noncompliance	LCO 3.0.2	YES	Same as above.

(a) The Applicable Safety Analyses 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[(a)]
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 new LCO 3.8.1.
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 2.
4.0.1	Operational Modes	SR 3.0.1	YES	Same as above.
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	5.5.6	YES	This Specification is actually a Surveillance Requirement which has been retained in the Administrative Controls programs for Inservice Testing.
4.0.6	Deleted by Amendments 243 (Unit 1) and 224 (Unit 2)	NA	NA	
4.0.7	Deleted by Amendments 243 (Unit 1) and 224 (Unit 2)	NA	NA	
4.0.8 (Unit 2 only)	Deleted by Amendment 224	NA	NA	
4.0.9 (Unit 2 only)	Deleted by Amendment 224	NA	NA	
3/4.1	**REACTIVITY CONTROL SYSTEMS**	**3.1**		
3/4.1.1	Boration Control			
3/4.1.1.1	Shutdown Margin - Tavg >200 F	3.1.1	YES-2	
3/4.1.1.2	Shutdown Margin - Tavg ≤ 200 F	3.1.1	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.

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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
3/4.1.1.3	Boron Dilution	Deleted	NO	Deleted, see Boron Dilution technical change discussion in the Discussion of Changes for CTS 3/4.1.1.3.
3/4.1.1.4	Moderator Temperature Coefficient	3.1.3	YES-2	
3/4.1.1.5	Minimum Temperature for Criticality	3.4.2	YES-2	
3/4.1.2	Boration Systems			
3/4.1.2.1	Flow Paths - Shutdown	Relocated	NO	See Appendix A, Page 1.
3/4.1.2.2	Flow Paths - Operating	Relocated	NO	See Appendix A, Page 3.
3/4.1.2.3	Charging Pump - Shutdown	Relocated	NO	See Appendix A, Page 5. The LCO 3.1.2.3.b requirements have been deleted. See Charging Pump - Shutdown technical change discussion in the Discussion of Changes for CTS 3/4.1.2.3. Asterisk requirement in LCO 3.1.2.3.b maintained in ITS 3.4.12.
3/4.1.2.4	Charging Pumps - Operating	Relocated	NO	See Appendix A, Page 6.
3/4.1.2.5	Boric Acid Transfer Pumps - Shutdown	Relocated	NO	See Appendix A, Page 1.
3/4.1.2.6	Boric Acid Transfer Pumps - Operating	Relocated	NO	See Appendix A, Page 3.
3/4.1.2.7	Borated Water Sources - Shutdown	Relocated	NO	See Appendix A, Page 7.
3/4.1.2.8	Borated Water Sources - Operations (Unit 1); Borated Water Sources- Operating (Unit 2)	Relocated	NO	See Appendix A, Page 9.
3/4.1.3	Movable Control Assemblies			
3/4.1.3.1	Group Height	3.1.4	YES-2	
3/4.1.3.2	Position Indicator Channels (Unit 1); Position Indicator Channels - Operating (Unit 2)	3.1.7	YES-2	
3/4.1.3.3 (Unit 2 only)	Deleted by Amendment 194	NA	NA	
3/4.1.3.3 (Unit 1); 3/4.1.3.4 (Unit 2)	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.
3/4.1.3.4 (Unit 1); 3/4.1.3.5 (Unit 2)	Shutdown Rod Insertion Limit	3.1.5	YES-2	
3/4.1.3.5 (Unit 1); 3/4.1.3.6 (Unit 2)	Control Rod Insertion Limits	3.1.6	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.

SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
3/4.2	**POWER DISTRIBUTION LIMITS**	**3.2**		
3/4.2.1	Axial Flux Difference	3.2.3	YES-2	
3/4.2.2	Heat Flux Hot Channel Factor - FQ(Z)	3.2.1	YES-2	
3/4.2.3	Nuclear Enthalpy Hot Channel Factor - $F_{\Delta H}^{(N)}$	3.2.2	YES-2	
3/4.2.4	Quadrant Power Tilt Ratio	3.2.4	YES-2	
3/4.2.5	DNB Parameters (Unit 1); DNB and Tavg Operating Parameters (Unit 2)	3.4.1	YES-2	
3/4.2.6	Allowable Power Level - APL	3.2.1	YES-2	
3/4.3	**INSTRUMENTATION**	**3.3**		
3/4.3.1	Reactor Trip System Instrumentation	3.3.1 3.3.8	YES-3	
3/4.3.2	Engineered Safety Feature Actuation System Instrumentation	3.3.2 3.3.5 3.3.6	YES-3	
3/4.3.3	Monitoring Instrumentation			
3/4.3.3.1	Radiation Monitoring Instrumentation			
Instrument 1.A.i	Area Monitors - Upper Containment	Deleted	NO	Deleted, see Radiation Monitoring Instrumentation technical change discussion in the Discussion of Changes for CTS 3/4.3.3.1.
Instrument 1.A.ii	Area Monitors - Containment High Range	3.3.3	YES-3	
Instrument 1.B	Process Monitors	3.4.15	YES-1	
Instrument 1.C	Noble Gas Effluent Monitors	Deleted	NO	Deleted, see Radiation Monitoring Instrumentation technical change discussion in the Discussion of Changes for CTS 3/4.3.3.1.
Instrument 2.A/B	MODE 6 Monitors	3.3.6	YES-3	
Instrument 3.A	Spent Fuel Storage	Deleted	NO	Deleted, see Radiation Monitoring Instrumentation technical change discussion in the Discussion of Changes for CTS 3/4.3.3.1.
3/4.3.3.2	Movable Incore Detectors	Relocated	NO	See Appendix A, Page 11.
3/4.3.3.3	Seismic Instrumentation	Relocated	NO	See Appendix A, Page 12.

(a) The Applicable Safety Analyses 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[a]
3/4.3.3.4	Meteorological Instrumentation	Relocated	NO	See Appendix A, Page 13.
3/4.3.3.5	Remote Shutdown Instrumentation	3.3.4	YES-4	
3/4.3.3.5.1	Appendix R Remote Shutdown Instrumentation	Relocated	NO	See Appendix A, Page 14.
3/4.3.3.6 (Unit 1); 3/4.3.3.7 (Unit 2)	Deleted by Amendments 120 (Unit 1) and 82 (Unit 2)	NA	NA	
3/4.3.3.7 (Unit 1); 3/4.3.3.8 (Unit 2)	Deleted by Amendments 208 (Unit 1) and 192 (Unit 2)	NA	NA	
3/4.3.3.8 (Unit 1); 3/4.3.3.6 (Unit 2)	Post-Accident Instrumentation	3.3.3	YES-3	See Appendix A, Page 15. Instrumentation that does not monitor Regulatory Guide 1.97 Type A or Category 1 variables has been relocated in accordance with the guidance provided in NUREG-1431, Revision 2.
3/4.3.3.9	Explosive Gas Monitoring Instrumentation	Relocated	NO	See Appendix A, Page 17.
3/4.4	**REACTOR COOLANT SYSTEM**	**3.4**		
3/4.4.1	Reactor Coolant Loops and Coolant Circulation			
3/4.4.1.1	Startup and Power Operation	3.4.4	YES-2	
3/4.4.1.2	Hot Standby	3.4.5	YES-3	
3/4.4.1.3	Hot Shutdown	3.4.6	YES-3	
3/4.4.1.4	Cold Shutdown - Loops Filled	3.4.7 3.4.12	YES-4 YES-2	
3/4.4.1.5	Cold Shutdown - Loops Not Filled	3.4.8	YES-4	
3/4.4.2	Safety Valves - Shutdown	3.4.10	YES-3	
3/4.4.3	Safety Valves - Operating	3.4.10	YES-3	
3/4.4.4	Pressurizer	3.4.9	YES-2	
3/4.4.5	Steam Generators	3.4.13 5.5	YES-2 YES	This Specification has been incorporated as a Surveillance Requirement (SR 3.4.13.2) in ITS 3.4.13 and a program in ITS 5.5.
3/4.4.6	Reactor Coolant System Leakage			
3/4.4.6.1	Leakage Detection Systems	3.4.15	YES-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.

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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[(a)]
3/4.4.6.2	Operational Leakage	3.4.13 3.4.14 3.5.5	YES-2	
3/4.4.7	Chemistry	Relocated	NO	See Appendix A, Page 18.
3/4.4.8	Specific Activity	3.4.16	YES-2	
3/4.4.9	Pressure/Temperature Limits			
3/4.4.9.1	Reactor Coolant System	3.4.3	YES-2	
3/4.4.9.2	Pressurizer	Relocated	NO	See Appendix A, Page 19.
3/4.4.9.3	Overpressure Protection Systems	3.4.12	YES-2	
3/4.4.10	Structural Integrity			
3/4.4.10.1	ASME Code Class 1, 2 and 3 Components	Relocated	NO	See Appendix A, Page 20. The Reactor Coolant Pump Flywheel Surveillance is being retained as a Program in ITS 5.5
3/4.4.11	Relief Valves - Operating	3.4.11	YES-3	
3/4.4.12	Reactor Coolant Vent System			
3/4.4.12.1	Reactor Vessel Head Vents	Relocated	NO	See Appendix A, Page 22.
3/4.4.12.2	Pressurizer Steam Space Vents	Relocated	NO	See Appendix A, Page 22.
3/4.5	**EMERGENCY CORE COOLING SYSTEMS**	**3.5**		
3/4.5.1	Accumulators	3.5.1	YES-3	
3/4.5.2	ECCS Subsystems - Tavg $\geq$ 350 F	3.5.2	YES-3	
3/4.5.3	ECCS Subsystems - Tavg $<$350 F	3.5.3 3.4.12	YES-3 YES-2	
3/4.5.4	Deleted by Amendments 158 (Unit 1) and 142 (Unit 2)			
3/4.5.5	Refueling Water Storage Tank	3.5.4	YES-3	
3/4.6	**CONTAINMENT SYSTEMS**	**3.6**		
3/4.6.1	Primary Containment			
3/4.6.1.1	Containment Integrity	3.6.1	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.

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SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
3/4.6.1.2	Containment Leakage	3.6.1	YES-3	Containment leakage is being retained as a Surveillance Requirement (SR 3.6.1.1) in ITS 3.6.1 and a program in ITS 5.5.16.
3/4.6.1.3	Containment Air Locks	3.6.2	YES-3	
3/4.6.1.4	Internal Pressure	3.6.4	YES-2	
3/4.6.1.5	Air Temperature	3.6.5	YES-2	
3/4.6.1.6	Containment 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.
3/4.6.1.7	Containment Ventilation System	3.6.3	YES-3	Containment purge valves are being retained as a Surveillance Requirement (SR 3.6.3.1) in ITS 3.6.3.
3/4.6.2	Depressurization and Cooling Systems			
3/4.6.2.1	Containment Spray System	3.6.6	YES-3	
3/4.6.2.2	Spray Additive Systems	3.6.7	YES-3	
3/4.6.3	Containment Isolation Valves	3.6.3	YES-3	
3/4.6.4	Combustible Gas Control			
3/4.6.4.1	Hydrogen Analyzers	3.3.3	YES-3	
3/4.6.4.2	Electric Hydrogen Recombiners-W	3.6.8	YES-3	
3/4.6.4.3	Distributed Ignition System	3.6.9	YES-4	
3/4.6.5	Ice Condenser			
3/4.6.5.1	Ice Bed	3.6.11	YES-3	
3/4.6.5.2	Ice Bed Temperature Monitoring System	Relocated	NO	See Appendix A, Page 23.
3/4.6.5.3	Ice Condenser Doors	3.6.12	YES-3	
3/4.6.5.4	Inlet Door Position Monitoring System	Relocated	NO	See Appendix A, Page 24.
3/4.6.5.5	Divider Barrier Personnel Access Doors and Equipment Hatches	3.6.13	YES-3	
3/4.6.5.6	Containment Air Recirculation Systems	3.6.10	YES-3	
3/4.6.5.7	Floor Drains	3.6.14	YES-3	
3/4.6.5.8	Refueling Canal Drains	3.6.14	YES-3	
3/4.6.5.9	Divider Barrier Seal	3.6.13	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.

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SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
3/4.7	**PLANT SYSTEMS**	**3.7**		
3/4.7.1	Turbine Cycle			
3/4.7.1.1	Safety Valves	3.7.1	YES-3	
3/4.7.1.2	Auxiliary Feedwater System	3.7.5	YES-3	
3/4.7.1.3	Condensate Storage System	3.7.6	YES-2, 3	
3/4.7.1.4	Activity	3.7.17	YES-2	
3/4.7.1.5	Steam Generator Stop Valves	3.7.2	YES-3	
3/4.7.2	Steam Generator Pressure/Temperature Limitation	Relocated	NO	See Appendix A, Page 25.
3/4.7.3	Component Cooling Water System	3.7.7	YES-3	
3/4.7.4	Essential Service Water System	3.7.8	YES-3	
3/4.7.5.1	Control Room Emergency Ventilation System	3.7.10 3.3.7	YES-3	
3/4.7.5.2	Control Room Air Conditioning System	3.7.11	YES-3	
3/4.7.6	ESF Ventilation System	3.7.12	YES-3	
3/4.7.7 (Unit 1); 3/4.7.8 (Unit 2)	Sealed Source Contamination	Relocated	NO	See Appendix A, Page 26.
3/4.7.8 (Unit 1); 3/4.7.7 (Unit 2)	Snubbers	Deleted	NO	Deleted, see Snubbers technical change discussion in the Discussion of Changes for CTS 3/4.7.8 (Unit 1) and 3/4.7.7 (Unit 2).
3/4.8	**ELECTRICAL POWER SYSTEM**	**3.8**		
3/4.8.1	A.C. Sources			
3/4.8.1.1	Operating	3.8.1 3.8.3	YES-3	
3/4.8.1.2	Shutdown	3.8.2 3.8.3	YES-3	
3/4.8.2	Onsite Power Distribution Systems			
3/4.8.2.1	A.C. Distribution - Operating	3.8.7 3.8.9	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.

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SUMMARY DISPOSITION MATRIX FOR CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[(a)]
3/4.8.2.2	A.C. Distribution - Shutdown	3.8.8 3.8.10	YES-3	
3/4.8.2.3	D.C. Distribution - Operating	3.8.4 3.8.6 3.8.9	YES-3	
3/4.8.2.4	D.C. Distribution - Shutdown	3.8.5 3.8.6 3.8.10	YES-3	
3/4.8.2.5	D.C. Distribution - Operating - Train N Battery System	3.8.4 3.8.6 3.8.9	YES-3	
3/4.8.3	Alternative A.C. Power Sources	Deleted	NO	Deleted, see Alternative AC Power Sources technical change discussion in the Discussion of Changes for CTS 3/4.8.3.1.
3/4.9	**REFUELING OPERATIONS**	**3.9**		
3/4.9.1	Boron Concentration	3.9.1	YES-2	
3/4.9.2	Instrumentation	3.9.2	YES-3	
3/4.9.3	Decay Time	Deleted	NO	Deleted, see Decay Time technical change discussion in the Discussion of Changes for CTS 3/4.9.3.
3/4.9.4	Containment Building Penetrations	3.9.3	YES-3	
3/4.9.5	Communications	Relocated	NO	See Appendix A, Page 27.
3/4.9.6	Deleted by Amendments 267 (Unit 1) and 248 (Unit 2)			
3/4.9.7	Deleted by Amendments 267 (Unit 1) and 248 (Unit 2)			
3/4.9.8	Residual Heat Removal and Coolant Circulation			
3/4.9.8.1	High Water Level	3.9.4 3.9.5	YES-4	
3/4.9.8.2	Low Water Level	3.9.5	YES-4	

(a) The Applicable Safety Analyses 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 CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[(a)]
3/4.9.9	Containment Purge and Exhaust Isolation System	3.3.6 3.9.3	YES-3	
3/4.9.10	Water Level - Reactor Vessel	3.9.6	YES-2	
3/4.9.11	Storage Pool Water Level	3.7.14	YES-2, 3	
3/4.9.12	Storage Pool Ventilation System	3.7.13	YES-3	
3/4.9.13	Spent Fuel Cask Movement	Deleted	NO	Deleted, see Spent Fuel Cask Movement technical change discussion in the Discussion of Changes for CTS 3/4.9.13.
3/4.9.14	Spent Fuel Cask Drop Protection System	Deleted	NO	Deleted, see Spent Fuel Cask Drop Protection System technical change discussion in the Discussion of Changes for CTS 3/4.9.14.
3/4.9.15	Storage Pool Boron Concentration	3.7.15	YES-2	
3/4.10	**SPECIAL TEST EXCEPTIONS**	**NA**		
3/4.10.1	Shutdown Margin	Deleted	NO	This Specification is provided to allow relaxation of the SDM LCO under certain specific conditions for testing. Direct application of the Technical Specification selection criteria is not appropriate. However, this special test exception is not necessary in the ITS since the SDM limit is in the COLR.
3/4.10.2	Group Height, Insertion and Power Distribution Limits	Deleted	NO	This Specification is provided to allow relaxation of certain LCOs under certain specific conditions for testing. Direct application of the Technical Specification selection criteria is not appropriate. However, this special test exception is not necessary in the ITS since the specific testing has been completed.
3/4.10.3 (Unit 1 only)	Pressure/Temperature Limitation - Reactor Criticality	Deleted	NO	Deleted, see Pressure/Temperature Limitation - Reactor Criticality technical change discussion in the Discussion of Changes for CTS 3/4.10.3.
3/4.10.4 (Unit 1); 3/4.10.3 (Unit 2)	Physics Tests	3.1.8	YES	This Specification is provided to allow relaxation of certain LCOs under certain specific conditions for testing. Direct application of the Technical Specification selection criteria is not appropriate. However, this special test exception is not necessary in the ITS since the specific testing has been completed.
3/4.10.5 (Unit 1); 3/4.10.4 (Unit 2)	Natural Circulation Tests (Unit 1); Reactor Coolant Loops (Unit 2)	Deleted	NO	Same as above.

(a) The Applicable Safety Analyses 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 CNP UNITS 1 AND 2

CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES[a]
3/4.10.5 (Unit 2 only)	Deleted by Amendment 194	NA	NA	
3/4.11	**RADIOACTIVE EFFLUENTS**	**NA**		
3/4.11.1	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			
3/4.11.2.1	Explosive Gas Mixture	5.5.10	YES	Same as above.
3/4.11.2.2	Gas Storage Tanks	5.5.10	YES	Same as above.
5.0	**DESIGN FEATURES**	**3.7.16** **4.0**	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.

(a) The Applicable Safety Analyses 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 SPECIFICATION RELOCATION

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.1 FLOW PATHS - SHUTDOWN
3/4.1.2.5 BORIC ACID TRANSFER PUMPS - SHUTDOWN

LCO STATEMENT:

3/4.1.2.1

As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid tanks via a boric acid transfer pump and charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if only the refueling water storage tank in Specification 3.1.2.7b is OPERABLE.

3/4.1.2.5

At least one boric acid transfer pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid transfer pump of Specification 3.1.2.1a is OPERABLE.

DISCUSSION:

The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the current specifications require a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event.

COMPARISON TO SCREENING CRITERIA:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The CVCS 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.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-6) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.1 FLOW PATHS - SHUTDOWN
3/4.1.2.5 BORIC ACID TRANSFER PUMPS - SHUTDOWN (continued)

CONCLUSION:

Since the screening criteria have not been satisfied, the Flow Paths - Shutdown LCO and Surveillances and the Boric Acid Transfer Pumps - Shutdown LCO and Surveillances may be relocated to other plant controlled documents outside Technical Specifications.

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APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.2 FLOW PATHS - OPERATING
3/4.1.2.6 BORIC ACID TRANSFER PUMPS - OPERATING

LCO STATEMENT:

3/4.1.2.2

Each of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System.

3/4.1.2.6

At least one boric acid transfer pump in the boron injection flow path required by Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

DISCUSSION:

The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the current specifications require a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event.

COMPARISON TO SCREENING CRITERIA:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The CVCS 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.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-8) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.2 FLOW PATHS - OPERATING
3/4.1.2.6 BORIC ACID TRANSFER PUMPS - OPERATING (continued)

CONCLUSION:

Since the screening criteria have not been satisfied, the Flow Paths - Operating LCO and Surveillances and the Boric Acid Transfer Pumps - Operating LCO and Surveillances may be relocated to other plant controlled documents outside Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.3 CHARGING PUMP - SHUTDOWN

LCO STATEMENT:

a. One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

DISCUSSION:

The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the current specifications require a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event. It should be noted that this LCO (part b) has requirements associated with the safe shutdown requirements of 10 CFR 50, Appendix R, and a requirement concerning the maximum number of charging and safety injection pumps that can be OPERABLE. These requirements are not covered by this discussion.

COMPARISON TO SCREENING CRITERIA:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The CVCS 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.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-6) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Charging Pump - Shutdown LCO and Surveillances may be relocated to other plant controlled documents outside Technical Specifications.

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APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.4 CHARGING PUMPS - OPERATING

LCO STATEMENT:

At least two charging pumps shall be OPERABLE.

DISCUSSION:

The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the current specifications require a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event.

COMPARISON TO SCREENING CRITERIA:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The CVCS 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.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-8) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Charging Pumps - Operating LCO and Surveillances may be relocated to other plant controlled documents outside Technical Specifications.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.7 BORATED WATER SOURCES - SHUTDOWN

LCO STATEMENT:

As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

   1. A minimum usable borated water volume of 5000 gallons,

   2. Between 6,550 and 6,990 ppm of boron, and

   3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

   1. A minimum usable borated water volume of 90,000 gallons,

   2. A minimum boron concentration of 2400 ppm, and

   3. A minimum solution temperature of 70°F.

DISCUSSION:

The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the current specifications require a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event.

COMPARISON TO SCREENING CRITERIA:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The CVCS 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.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-10) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.7 BORATED WATER SOURCES - SHUTDOWN (continued)

CONCLUSION:

Since the screening criteria have not been satisfied, the Borated Water Sources - Shutdown LCO and Surveillances may be relocated to other plant controlled documents outside Technical Specifications.

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APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.8 BORATED WATER SOURCES - OPERATIONS (UNIT 1)/OPERATING (UNIT 2)

LCO STATEMENT:

Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

   1. A minimum contained borated water volume of 8,500 gallons,

   2. Between 6,550 and 6,990 ppm of boron, and

   3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

   1. A minimum contained borated water volume of 375,500 gallons of water,

   2. Between 2400 and 2600 ppm of boron, and

   3. A minimum solution temperature of 70°F and a maximum solution temperature of 100°F.

DISCUSSION:

The boration subsystem of the Chemical and Volume Control System (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN. To accomplish this functional requirement, the current specifications require a source of borated water, one or more flow paths to inject this borated water into the RCS, and appropriate charging pumps to provide the necessary charging head.

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the CVCS that causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system. This action is required before the SHUTDOWN MARGIN is lost. Operation of the boration subsystem is not assumed to mitigate this event.

COMPARISON TO SCREENING CRITERIA:

1. The CVCS is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The CVCS 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.

3. The CVCS is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-10) and summarized in Table 1 of WCAP-11618, the loss of the CVCS was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.1.2.8 BORATED WATER SOURCES - OPERATIONS (UNIT 1)/OPERATING (UNIT 2) (continued)

CONCLUSION:

Since the screening criteria have not been satisfied, the Borated Water Sources - Operations/Operating LCO and Surveillances may be relocated to other plant controlled documents outside Technical Specifications.

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APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.3.3.2 MOVABLE INCORE DETECTORS

LCO STATEMENT:

The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

DISCUSSION:

This Specification ensures the OPERABILITY of Movable Incore Detector Instrumentation when required to monitor the flux distribution within the core. The System 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 DBA analysis and does not mitigate an accident.

COMPARISON TO SCREENING CRITERIA:

1. This system is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. This system is not a process variable that is an initial condition in a DBA or transient analyses.

3. This system does not act as a part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-12) and summarized in Table 1 of WCAP-11618, the loss of Movable Incore Detectors was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Movable Incore Detectors LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

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APPENDIX A
JUSTIFICATION FOR SPECIFICATION RELOCATION

3/4.3.3.3 SEISMIC INSTRUMENTATION

LCO STATEMENT:

The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

DISCUSSION:

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 DBA.

COMPARISON TO SCREENING CRITERIA:

1. These instruments are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. These instruments do not monitor a process variable that is an initial condition of a DBA or transient analyses.

3. These instruments are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-22), and summarized in Table 1 of WCAP-11618, the loss of seismic monitoring instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

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

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3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

LCO STATEMENT:

The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

DISCUSSION:

Meteorological instrumentation is used to measure environmental parameters that may affect distribution of fission products and gases following a design basis accident (DBA), but it is not an input assumption for any DBA analysis and does not mitigate the accident. Meteorological information is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

COMPARISON TO SCREENING CRITERIA:

1. These instruments are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. These instruments do not monitor a process variable that is an initial condition of a DBA or transient analyses.

3. These instruments are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-23), and summarized in Table 1 of WCAP-11618, the loss of meteorological monitoring instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

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

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3/4.3.3.5.1 APPENDIX R REMOTE SHUTDOWN INSTRUMENTATION

LCO STATEMENT:

The Appendix R remote shutdown instrumentation channels shown in Table 3.3-9A shall be OPERABLE with an opposite unit power supply available and with read out capability at the LSI panels.

DISCUSSION:

The Appendix R Remote Shutdown Instrumentation is used to ensure that a fire will not preclude achieving safe shutdown. This instrumentation is independent of areas where a fire could damage systems normally used to shutdown the reactor. However, the instrumentation is not used to detect a degradation of the reactor coolant pressure boundary, and is not assumed to mitigate a design basis accident (DBA) or transient event. The Appendix R Remote Shutdown Instrumentation capability is consistent with the requirements of 10 CFR 50, Appendix R. The acceptability of the relocation of the Appendix R Technical Specification requirements from the plant Technical Specifications has already been endorsed by the NRC as indicated in Generic Letter 86-10.

COMPARISON TO SCREENING CRITERIA:

1. The Appendix R Remote Shutdown Instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The Appendix R Remote Shutdown Instrumentation does not monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The Appendix R Remote Shutdown Instrumentation is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Appendix B, page 1, I&M found the loss of the Appendix R Remote Shutdown Instrumentation to be a non-significant risk contributor to core damage frequency and offsite releases.

CONCLUSION:

Since the screening criteria have not been satisfied, the Appendix R Remote Shutdown Instrumentation LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

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3/4.3.3.8 (Unit 1); POST-ACCIDENT INSTRUMENTATION
3/4.3.3.6 (Unit 2)

LCO STATEMENT:

The post-accident monitoring instrumentation channels shown in Table 3.3-11 (Unit 1) and 3.3-10 (Unit 2) shall be OPERABLE.

DISCUSSION:

Each individual 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 deterministic screening criteria to post-accident monitoring instrumentation is documented in letter dated May 9, 1988 from T.E. Murley (NRC) to W.S. Wilgus (NRC Split Report to Owners Groups). The 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 instruments specified in the plant's Safety Evaluation Report (SER) on Regulatory Guide 1.97, and all Regulatory Guide 1.97 Category 1 instruments. Accordingly, this position has been applied to the CNP Units 1 and 2 Regulatory Guide 1.97 instruments. Those instruments meeting these criteria have remained in Technical Specifications. The instruments not meeting this criteria will be relocated from the Technical Specifications to plant controlled documents.

A review of the CNP Units 1 and 2 UFSAR and the NRC Regulatory Guide 1.97 Safety Evaluation for CNP Units 1 and 2 shows that the following Unit 1 CTS Tables 3.3-11 and 4.3-7 and Unit 2 CTS Tables 3.3-10 and 4.3-10 Instruments do not meet Category 1 or Type A requirements.

Instrument 9 Boric Acid Tank Solution Level
Instrument 12 PORV Position Indicator - Limit Switches
Instrument 13 PORV Block Valve Position Indicator - Limit Switches
Instrument 14 Safety Valve Position Indicator - Acoustic Monitor
Instrument 17 Containment Sump Level

COMPARISON TO SCREENING CRITERIA:

1. These instruments are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The monitored parameters are not process variables, design features, or operating restrictions that are initial conditions of a DBA or transient.

3. These instruments are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-25) and summarized in Table 1 of WCAP-11618, the loss of the (above listed) instruments were found to be non-significant risk contributors to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

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3/4.3.3.8 (Unit 1); POST-ACCIDENT INSTRUMENTATION (continued)
3/4.3.3.6 (Unit 2)

CONCLUSION:

Since the screening criteria have not been satisfied for instruments which do not meet Regulatory Guide 1.97 Type A variable requirements or Category 1 variable requirements, their associated LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

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3/4.3.3.9 EXPLOSIVE GAS MONITORING INSTRUMENTATION

LCO STATEMENT:

The explosive gas monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded.

DISCUSSION:

The explosive gas monitor Specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the gaseous waste processing system is adequately monitored, which will help ensure that the concentration is maintained below the flammability limit. However, the system is designed to contain detonations, and detonations would not affect the function of any safety related equipment. The concentration of oxygen in the gaseous Waste Processing System is not an initial assumption of any design basis accident (DBA) or transient analysis.

COMPARISON TO SCREENING CRITERIA:

1. The 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. The explosive gas monitoring instrumentation is not used to monitor a process variable 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. The 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. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

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

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3/4.4.7 CHEMISTRY

LCO STATEMENT:

The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

DISCUSSION:

Poor coolant water chemistry contributes to the long term degradation of system materials of construction, and thus is not of immediate importance to the unit operator. Reactor coolant water chemistry is monitored for a variety of reasons. One reason is to reduce the possibility of failures in the Reactor Coolant System pressure boundary caused by corrosion. However, the chemistry monitoring activity is of a long term preventative purpose rather than mitigative.

COMPARISON TO SCREENING CRITERIA:

1. Reactor coolant water chemistry is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Reactor coolant water chemistry is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. Reactor coolant water chemistry is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-40) and summarized in Table 1 of WCAP-11618, the reactor coolant water chemistry was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

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

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3/4.4.9.2 PRESSURIZER

LCO STATEMENT:

The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,
b. A maximum cooldown of 200°F in any one hour period, and
c. A maximum spray water temperature differential of 320°F.

DISCUSSION:

The heatup and cooldown rate limits and spray water differential limit are placed on the pressurizer to prevent non-ductile failure and assure compatibility of operation with the fatigue analysis performed. The limits meet the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G. These limitations are consistent with structural analysis results. However, these limits are not initial condition assumptions of a DBA or transient. These limits represent operating restrictions and Criterion 2 includes operating restrictions. However, it should be noted that in the Final Policy Statement the Criterion 2 discussion specified only those operating restrictions required to preclude unanalyzed accidents and transients be included in Technical Specifications.

COMPARISON TO SCREENING CRITERIA:

1. Pressurizer heatup and cooldown limits and spray water differential limit are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Pressurizer heatup and cooldown limits and spray water differential limit are not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. Pressurizer heatup and cooldown limits and spray water differential limit are not a part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-41) and summarized in Table 1 of WCAP-11618, the pressurizer heatup and cooldown limits and spray water differential limit were found to be non-significant risk contributors to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

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

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3/4.4.10.1 STRUCTURAL INTEGRITY - ASME CODE CLASS 1, 2 AND 3 COMPONENTS

LCO STATEMENT:

The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

DISCUSSION:

The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained throughout the life of the component. ASME Code Class 1, 2, and 3 components are monitored so that the possibility of component structural failure does not degrade the safety function of the system. The monitoring activity is of a preventive nature rather than a mitigative action. Other Technical Specifications require important systems to be OPERABLE (for example, Emergency Core Cooling Systems) and in a ready state for mitigative action. This Technical Specification is more directed toward prevention of component degradation and continued long term maintenance of acceptable structural conditions. Hence, it is not necessary to retain this Specification to ensure immediate OPERABILITY of safety systems.

Further, this Technical Specification prescribes inspection requirements that are performed during plant shutdown. It is, therefore, not directly important for responding to design basis accidents.

COMPARISON TO SCREENING CRITERIA:

1. The inspections stipulated by this Specification are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary during operations prior to a design basis accident (DBA).

2. The inspections stipulated by this Specification are not a process variable, design feature, or operating restriction that is an initial assumption in a DBA or transient.

3. The ASME Code Class 1, 2, and 3 components inspected per this Specification are assumed to function to mitigate a DBA. Their capability to perform this function is addressed by other Technical Specifications. This Technical Specification only specifies inspection requirements for these components, and these inspections can only be performed when the plant is shutdown. Therefore, Criterion 3 is not satisfied.

4. As discussed in Section 4.0 (Appendix A, page A-43) and summarized in Table 1 of WCAP-11618 the assurance of operability of the entire system as verified in the system operability Specification dominates the risk contribution of the system. The lack of a long term assurance of structural integrity as stipulated by this Specification was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Structural Integrity - ASME Code Class 1, 2, and 3 Components LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications. In addition, surveillances, except for the reactor coolant pump (RCP) flywheel inspection, are already required by regulations in 10 CFR 50.55a to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda.

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3/4.4.10.1 STRUCTURAL INTEGRITY - ASME CODE CLASS 1, 2 AND 3 COMPONENTS (continued)

The RCP flywheel inspection requirement is not covered by other regulatory requirements and is needed for safe operation of the plant; therefore, this requirement will be maintained in the CNP Units 1 and 2 Improved Technical Specifications. Chapter 5.0 of the CNP Units 1 and 2 Improved Technical Specifications will contain a section which provides a programmatic approach to the requirements relating to the structural integrity of ASME Code Class 1, 2, and 3 components.

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3/4.4.12.1 REACTOR VESSEL HEAD VENTS
3/4.4.12.2 PRESSURIZER STEAM SPACE VENTS

LCO STATEMENT:

3/4.4.12.1

At least one of the Reactor Vessel head vent paths, consisting of two remotely operated valves in series, powered from Class 1E DC busses, shall be OPERABLE and closed.

3/4.4.12.2

At least one of the pressurizer steam space vent paths, each consisting of two remotely operated valves in series, powered from Class 1E DC busses, shall be OPERABLE and closed.

DISCUSSION:

The reactor vessel head and pressurizer steam space vents are provided to exhaust noncondensible gases and/or steam from the RCS which could inhibit natural circulation core cooling following any event involving a loss of offsite power and requiring long term cooling, such as a loss-of-coolant accident (LOCA). Their function, capabilities, and testing requirements are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," however, the operation of reactor vessel head vents is not part of the primary success path. The operation of these vents is an operator action after the event has occurred, and is only required when there is indication that natural circulation is not occurring.

COMPARISON TO SCREENING CRITERIA:

1. Reactor vessel head and pressurizer steam space vents are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Reactor vessel head and pressurizer steam space vents are not process variables, design features, or operating restrictions that are an initial condition of a DBA or transient.

3. Reactor vessel head and pressurizer steam space vents are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-44) and summarized in Table 1 of WCAP-11618, the reactor vessel head and pressurizer steam space vents were found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Reactor Vessel Head Vents LCO and Surveillances and Pressurizer Steam Space Vents LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

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3/4.6.5.2 ICE BED TEMPERATURE MONITORING SYSTEM

LCO STATEMENT:

The ice bed temperature monitoring system shall be OPERABLE with at least 2 OPERABLE RTD channels in the ice bed at elevations 652' 2 1/4", 672' 5 1/4" and 696' 2 1/4" for each one third of the ice condenser.

DISCUSSION:

The Ice Bed Temperature Monitoring System monitors the temperature of the ice bed to ensure that the ice bed temperature does not increase above the required limits undetected. However, the Ice Bed Temperature Monitoring System is not required to ensure the ice bed temperature is maintained within limits. Another Technical Specification (that is being retained) will continue to ensure that temperature is maintained within the required limits.

COMPARISON TO SCREENING CRITERIA:

1. The Ice Bed Temperature Monitoring System is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The Ice Bed Temperature Monitoring System is are not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. The Ice Bed Temperature Monitoring System is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-78) and summarized in Table 1 of WCAP-11618, the Ice Bed Temperature Monitoring System was found to be non-significant risk contributors to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Ice Bed Temperature Monitoring System LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

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3/4.6.5.4 INLET DOOR POSITION MONITORING SYSTEM

LCO STATEMENT:

The inlet door position monitoring system shall be OPERABLE.

DISCUSSION:

The Inlet Door Position Monitoring System monitors the position of the ice bed inlet doors during normal operation to ensure that the ice bed inlet doors do not open (which could allow the ice bed temperature to increase above the required limits). However, the Inlet Door Position Monitoring System is not required to ensure the inlet doors remain closed and ice bed temperature is maintained within limits. Other Technical Specifications (that are being retained) will continue to ensure that the inlet doors remain closed and temperature is maintained within the required limits.

COMPARISON TO SCREENING CRITERIA:

1. The Inlet Door Position Monitoring System is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The Inlet Door Position Monitoring System is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. The Inlet Door Position Monitoring System is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-78) and summarized in Table 1 of WCAP-11618, the Inlet Door Position Monitoring System was found to be non-significant risk contributors to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Inlet Door Position Monitoring System LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

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3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LCO STATEMENT:

The temperatures of both the primary and secondary coolants in the steam generators shall be > 70°F when the pressure of either coolant in the steam generator is > 200 psig.

DISCUSSION:

The limitation on steam generator pressures and temperatures ensures that pressure-induced stresses on the steam generators do not exceed the maximum allowable fracture toughness limits. These pressure and temperature limits are based on maintaining a steam generator RT_{NDT} sufficient to prevent brittle fracture. As such, the Technical Specification places limits on variables consistent with structural analysis results. However, these limits are not initial condition assumptions of a DBA or transient. These limits represent operating restrictions and Criterion 2 includes operating restrictions. However, it should be noted that in the Final Policy Statement the Criterion 2 discussion specified only those operating restrictions required to preclude unanalyzed accidents and transients be included in Technical Specifications.

COMPARISON TO SCREENING CRITERIA:

1. The steam generator P/T limits are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The steam generator P/T limits are not process variables, design features, or operating restrictions that are an initial condition of a DBA or transient.

3. The steam generator P/T limits are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-55) and summarized in Table 1 of WCAP-11618, the steam generator P/T limits were found to be non-significant risk contributors to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Steam Generator P/T Limitation LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

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3/4.7.7 (Unit 1); SEALED SOURCE CONTAMINATION
3/4.7.8 (Unit 2)

LCO STATEMENT:

Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material, shall be free of 0.005 microcuries of removable contamination.

DISCUSSION:

The limitations on sealed source contamination are intended to ensure that the total body and individual organ irradiation doses do not exceed allowable limits in the event of ingestion or inhalation. This is done by imposing a maximum limitation of $\leq$ 0.005 microcuries of removable contamination on each sealed source. This requirement and the associated surveillance requirements bear no relation to the conditions or limitations that are necessary to ensure safe reactor operation.

COMPARISON TO SCREENING CRITERIA:

1. Sealed source contamination is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Sealed source contamination is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. Sealed source contamination is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-59) and summarized in Table 1 of WCAP-11618, the sealed source contamination being not within limits was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Sealed Source Contamination LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

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3/4.9.5 COMMUNICATIONS

LCO STATEMENT:

Direct communications shall be maintained between the control room and personnel at the refueling station.

DISCUSSION:

Communication between the control room personnel and personnel performing CORE ALTERATIONS is maintained to ensure that personnel can be promptly informed of significant changes in the plant status or core reactivity condition during refueling. The communications allow for coordination of activities that require interaction between the control room and containment personnel. The prompt notification of the control room of a fuel handling accident is not an assumption in the fuel handling accident analysis. While notification is necessary to ensure the control room is isolated to meet the control room operator dose limits in General Design Criteria 19, the fuel handling accident analysis does not take credit for direct communications between the refueling station and the control room (30 minutes is assumed before control room operators isolate the control room).

COMPARISON TO SCREENING CRITERIA:

1. Communications during refueling operations is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Communications during refueling operations is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. Communication during refueling operations is not a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-67) and summarized in Table 1 of WCAP-11618, the loss of communications was found to be a non-significant risk contributor to core damage frequency and offsite releases. I&M has reviewed this evaluation, considers it applicable to CNP Units 1 and 2, and concurs with the assessment.

CONCLUSION:

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

APPENDIX B

COOK NUCLEAR PLANT UNITS 1 AND 2
SPECIFIC RISK SIGNIFICANT EVALUATIONS

APPENDIX B
COOK NUCLEAR PLANT UNITS 1 AND 2
SPECIFIC RISK SIGNIFICANT EVALUATIONS

TECHNICAL SPECIFICATION 3/4.3.3.5.1 APPENDIX R REMOTE SHUTDOWN INSTRUMENTATION

DESCRIPTION OF REQUIREMENT:

The Appendix R remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown of the facility to a COLD SHUTDOWN condition in the event of a fire in the main control room. This Specification maintains this requirement.

POTENTIAL EFFECT:

Loss of capability to place a unit in COLD SHUTDOWN as a result of a fire in the main control room.

REFERENCED DOCUMENTS UTILIZED:

1. CNP Updated Final Safety Analysis Report, Section 7.7, Operating Control Stations.

2. CNP Technical Specifications, Facility Operating Licenses DPR-58 and DPR-74.

3. CNP Probabilistic Risk Assessment Final Report, Volume 11.

4. 10 CFR Part 50, Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979.

5. NRC Generic Letter 86-10, Implementation of Fire Protection Requirements.

COMMENTS:

Although the Appendix R remote shutdown instrumentation has not been specifically evaluated for risk significance either generically or on a plant specific basis, insight based on a review of the referenced documents indicates that the instrumentation is not risk dominant with regards to core damage frequency or off-site health effects. Furthermore, Generic Letter 86-10 identifies conditions under which fire protection related Technical Specifications may be relocated to other administratively controlled documents, subject to the provisions of 10 CFR 50.59, without a significant increase to public health and safety.

CONCLUSION:

Based on a thorough review of the listed references, it is recommended that CNP Technical Specification 3/4.3.3.5.1 be relocated from Technical Specifications to a licensee controlled document.