IMPROVED TECHNICAL SPECIFICATIONS

MONTICELLO NUCLEAR GENERATING PLANT

VOLUMES 1 – 5
REVISION 1

1. Application of Selection Criteria to the MNGP Tech Specs
2. Generic Determination of NSHC and Environmental Assess
3. ITS Chapter 1.0, - Use and Application
4. ITS Chapter 2.0, - Safety Limits
5. ITS Section 3.0, - LCO Applicability and SR Applicability

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Summary of Changes
Application of Selection Criteria to the
Monticello Technical Specifications

No changes are required for the Volume.

ATTACHMENT 1

VOLUME 1

MONTICELLO
IMPROVED TECHNICAL
SPECIFICATIONS CONVERSION

APPLICATION OF SELECTION
CRITERIA TO THE
MONTICELLO
TECHNICAL SPECIFICATIONS

Revision 1

APPLICATION OF SELECTION CRITERIA TO THE MONTICELLO TECHNICAL SPECIFICATIONS

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APPENDIX

APPLICATION OF SELECTION CRITERIA TO THE MONTICELLO TECHNICAL SPECIFICATIONS

1. INTRODUCTION

The purpose of this document is to confirm the results of the BWR Owners Group application of the Technical Specification selection criteria on a plant specific basis for Monticello Nuclear Power Station. Nuclear Management Company, LLC has reviewed the application of the selection criteria to each of the Technical Specifications utilized in BWROG report NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," including Supplement 1 (Reference 1), NUREG-1433, Standard Technical Specifications, General Electric Plants, BWR/4," (Reference 2) and applied the criteria to each of the current Monticello Technical Specifications. Additionally, in accordance with the NRC guidance, this confirmation of the application of selection criteria to Monticello includes confirming the risk insights from Probabilistic Risk Assessment (PRA) evaluations, provided in Reference 1, as applicable to Monticello.

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APPLICATION OF SELECTION CRITERIA
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2. SELECTION CRITERIA

Nuclear Management Company, LLC 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. Probabilistic Risk Assessment (PRA) insights as used in the BWROG submittal were utilized, confirmed by Nuclear Management Company, LLC 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).

**APPLICATION OF SELECTION CRITERIA
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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:

APPLICATION OF SELECTION CRITERIA TO THE MONTICELLO TECHNICAL SPECIFICATIONS

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.

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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. Those Technical Specifications proposed to remain part of the Improved Technical Specifications were not reviewed. This review was accomplished in Reference 1 except where discussed in Appendix A, "Justification For Specification Relocation," and has been confirmed by Nuclear Management Company, LLC for those Specifications to be relocated. The Monticello plant-specific Probabilistic Risk Assessment (PRA) was reviewed during this process.

Assumptions and Approach

Briefly, the approach used in Reference 1 was the following:

The risk assessment analysis evaluated the loss of function of the system or component whose LCO was being considered for relocation and qualitatively assessed the associated effect on core damage frequency and offsite releases. The assessment was based on available literature on plant risk insights and PRAs. Table 3-1 lists the PRAs used for making the assessments and is provided at the end of this section. A detailed quantitative calculation of the core damage and offsite release effects was not performed. However, the analysis did provide an indication of the relative significance of those LCOs proposed for relocation on the likelihood or severity of the accident sequences that are commonly found to dominate plant safety risks. The following analysis steps were performed for each LCO proposed for relocation:

a. List the function(s) affected by removal of the LCO item.

b. Determine the effect of loss of the LCO item on the function(s).

c. Identify compensating provisions, redundancy, and backups related to the loss of the LCO item.

d. Determine the relative frequency (high, medium, and low) of the loss of the function(s) assuming the LCO item is removed from Technical Specifications and controlled by other procedures or programs. Use information from current PRAs and related analyses to establish the relative frequency.

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3. PRA INSIGHTS (continued)

e. Determine the relative significance (high, medium, and low) of the loss of the function(s). Use information from current PRAs and related analyses to establish the relative significance.

f. Apply risk category criteria to establish the potential risk significance or non-significance of the LCO item. Risk categories were defined as follows:

RISK CRITERIA

	Consequence		
Frequency	High	Medium	Low
High	S	S	NS
Medium	S	S	NS
Low	NS	NS	NS

S = Potential Significant Risk Contributor

NS = Risk Non-Significant

g. List any comments or caveats that apply to the above assessment. The output from the above evaluation was a list of LCOs proposed for relocation that could have potential plant safety risk significance if not properly controlled by other procedures or programs. As a result these Specifications will be relocated to other plant controlled documents outside the Technical Specifications.

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TABLE 3-1

BWR PRAs USED IN NEDO-31466 (and Supplement 1)
RISK ASSESSMENT

- BWR/6 Standard Plant, GESSAR II, 238 Nuclear Island, BWR/6 Standard Plant Probabilistic Risk Assessment, Docket No. STN 50-447, March 1982.
- La Salle County Station, NEDO-31085, Probabilistic Safety Analysis, February 1988.
- Grand Gulf Nuclear Station, IDCOR, Technical Report 86.2GG, Verification of IPE for Grand Gulf, March 1987.
- Limerick, Docket Nos. 50-352, 50-353, 1981, "Probabilistic Risk Assessment, Limerick Generating Station," Philadelphia Electric Company.
- Shoreham, Probabilistic Risk Assessment Shoreham Nuclear Power Station, Long Island Lighting Company, SAI-372-83-PA-01, June 24, 1983.
- Peach Bottom 2, NUREG-75/0104, "Reactor Safety Study," WASH-1400, October 1975.
- Millstone Point 1, NUREG/CR-3085, "Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 1 Nuclear Power Plant," January 1983.
- Grand Gulf, NUREG/CR-1659, "Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant," October 1981.
- NEDC-30936P, "BWR Owners' Group Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation) Part 2," June 1987.

APPLICATION OF SELECTION CRITERIA
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4. RESULTS OF APPLICATION OF SELECTION CRITERIA

The selection criteria from Section 2 were applied to the Monticello Technical Specifications. The attachment is a summary of that application indicating which Specifications are being retained or relocated. Discussions that document the rationale for the relocation of each Specification which failed to meet the selection criteria are provided in Appendix A. No Significant Hazards Considerations (10 CFR 50.92) evaluations for those Specifications relocated are provided with the Discussion of Changes for the specific Technical Specifications. Nuclear Management Company, LLC will relocate those Specifications identified as not satisfying the criteria to licensee controlled documents whose changes are governed by 10 CFR 50.59.

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5. REFERENCES

1. NEDO-31466 (and Supplement 1), "Technical Specification Screening Criteria Application and Risk Assessment," November 1987 and July 1989.

2. NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," Revision 3, June 2004.

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

ATTACHMENT 1

SUMMARY DISPOSITION MATRIX
FOR
MONTICELLO

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.A	Reactor Core Safety Limits	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	Reactor Coolant System Pressure Safety Limit	2.1.2	YES	Same as above.
2.2	Safety Limit Violations	2.2	YES	Same as above.
3/4.0	SURVEILLANCE REQUIREMENTS - APPLICABILITY	3.0		
4.0.A	Meeting Surveillance Requirements and Time of Performance	SR 3.0.1	YES	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements of 4.0 will be retained in Technical Specifications, as modified consistent with NUREG-1433, Revision 3.
4.0.B	Time Interval Extensions	SR 3.0.2	YES	Same as above.
4.0.C	Noncompliance and Time of Performance	SR 3.0.1, SR 3.0.4	YES	Same as above.
4.0.D	Missed Surveillances	SR 3.0.1	YES	Same as above.
4.0.E	Delay Time for Missed Surveillances	SR 3.0.3	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)
3/4.1	**REACTOR PROTECTION SYSTEM**			
3/4.1.A and B	Reactor Protection System Instrumentation	1.1, 3.3.1.1, 3.3.6.1, 3.3.6.2	YES-3	
3/4.1.A and B and Table 3.1.1 Trip Function 9, Table 4.1.1 Instrument Channel 5, and Table 4.1.2 Instrument Channel 7	Turbine Condenser Low Vacuum	Relocated	NO	See Appendix A, page 1.
3/4.1.C	RPS Power Monitoring System	3.3.8.2	YES-3	
3/4.2	**PROTECTIVE INSTRUMENTATION**	**3.3**		
3/4.2.A	Primary Containment Isolation Functions	3.3.6.1	YES-3, 4	
3/4.2.B	Emergency Core Cooling Subsystems Actuation	3.3.5.1, 3.3.8.1	YES-3	
3/4.2.C	Control Rod Block Actuation			
3/4.2.C.1	SRM, IRM, APRM and Scram Discharge Volume Rod Blocks	Relocated	NO	See Appendix A, pages 2 through 5.
3/4.2.C.2	Rod Block Monitor	3.3.2.1	YES-3	
3/4.2.D	Other Instrumentation	3.3.5.1, 3.3.5.2	YES -3, 4	
3/4.2.E	Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation	3.3.6.2	YES-3	
3/4.2.F	Recirculation Pump Trip and Alternate Rod Injection Initiation	3.3.4.1	YES-4	
3/4.2.G	Safeguards Bus Voltage Protection	3.3.8.1	YES-3	
3/4.2.H	Instrumentation for Safety/Relief Valve Low-Low Set Logic	3.3.6.3, 3.6.1.5	YES-3	
3/4.2.I	Instrumentation for Control Room Habitability Protection	3.3.7.1	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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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3/4.3	**CONTROL ROD SYSTEM**	3.1		
3/4.3.A	Reactivity Limitations			
3/4.3.A.1	Reactivity margin - core loading	1.1, 3.1.1	YES-2	
3/4.3.A.2	Reactivity margin - stuck control rods	3.1.3	YES-3	
3/4.3.B	Control Rod Withdrawal			
3/4.3.B.1	Coupling	3.1.3, 3.10.5	YES-3	
3/4.3.B.2	Control Rod Drive Housing Support System	Deleted	NO	Deleted, see CRD Housing Support technical change discussion in the Discussion of Changes for CTS: 3/4.3.B.2
3/4.3.B.3.(a)	Control Rod Withdrawal Sequences	3.1.6, 3.3.2.1	YES-3	
3/4.3.B.3.(b)	Rod Worth Minimizer	3.3.2.1	YES-3	
3/4.3.B.4	Source Range Monitors for startup and refueling	3.3.1.2	YES	
3/4.3.C	Scram Insertion Times	3.1.4	YES-3	
3/4.3.D	Control Rod Accumulators	3.1.5, 3.9.5	YES-3	
3/4.3.E	Reactivity Anomalies	3.1.2	YES-2	
3/4.3.F	Scram Discharge Volume	3.1.8	YES-3	
3/4.3.G	Required Action	3.1.1, 3.1.3, 3.1.4, 3.1.5, 3.1.6, 3.3.1.2, 3.9.5	YES	This requirement provides the appropriate actions to take if CTS 3.3.A through D are not met. As such, direct application of the Technical Specification selection criteria is not appropriate for actions. Therefore, changes to this action are discussed in the technical change discussion in the Discussion of Changes for ITS 3.1.1, 3.1.3, 3.1.4, 3.1.5, 3.1.6, 3.3.1.2, and 3.9.5.
3/4.4	**STANDBY LIQUID CONTROL SYSTEM**			
3/4.4.A	Standby Liquid Control System	3.1.7	YES-4	
3/4.4.B	Boron Solution Requirements	3.1.7	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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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3/4.4.C	Required Action	3.1.7	YES	This requirement provides the appropriate action to take if CTS 3.4.A or B is not met. As such, direct application of the Technical Specification selection criteria is not appropriate for actions. Therefore, changes to this action are discussed in the technical change discussion in the Discussion of Changes for ITS 3.1.7.
3/4.5	**CORE AND CONTAINMENT SPRAY/COOLING SYSTEMS**	**3.5**		
3/4.5.A	ECCS Systems	3.5.1, 3.10.1	YES-3	
4.5.A.4	ADS Inhibit Switch	Relocated	NO	See Appendix A, page 6.
3/4.5.B	RHR Intertie Return Line Isolation Valves	3.5.1	YES-3	
3/4.5.C	Containment Spray/Cooling System	3.6.2.3, 3.6.1.8, 3.7.1	YES-3	
3/4.5.D	RCIC	3.3.5.2, 3.5.3, 3.10.1	YES-4	
3/4.5.E	Cold Shutdown and Refueling Requirements	3.5.2	YES-3	
3/4.5.F	Recirculation System	3.4.1	YES-2	
3/4.6	**PRIMARY SYSTEM BOUNDARY**	**3.4**		
3/4.6.A	Reactor Coolant Heatup and Cooldown	3.4.9	YES-2	
3/4.6.B	Reactor Vessel Temperature and Pressure	3.4.9	YES-2	
3/4.6.C	Coolant Chemistry			
3/4.6.C.1	Radioiodine concentration in the reactor coolant	3.4.6, 3.10.1	YES-2	
3/4.6.C.2 and 3	Reactor Coolant Water Chemistry	Relocated	NO	See Appendix A, page 7.

(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)
3/4.6.C.4	Required Action	3.4.6	YES	This requirement provides the appropriate actions to take if CTS 3.6.C.1 through 3 are not met. As such, direct application of the Technical Specification selection criteria is not appropriate for actions. Therefore, changes to this action are discussed in the technical change discussion in the Discussion of Changes for ITS 3.4.6 and CTS 3/4.6.C.2 and 3.
3/4.6.D	Reactor Coolant System (RCS)			
3/4.6.D.1	Operational Leakage	3.4.4	YES-2	
3/4.6.D.2	RCS Leakage Detection Instrumentation	3.4.5	YES-1	
3/4.6.E	Safety/Relief Valves	3.4.3, 3.6.1.5	YES-3	
3/4.6.F	Deleted by Amendent 42			
3/4.6.G	Jet Pumps	3.4.2	YES-2	
3/4.6.H	Snubbers	Deleted	NO	Deleted, see Snubbers technical change discussion in the Discussion of Changes for CTS: 3/4.6.H.
3/4.7	**CONTAINMENT SYSTEMS**	**3.6**		
3/4.7.A	Primary Containment			
3/4.7.A.1	Suppression Pool Volume and Temperature	3.5.2, 3.6.1.1, 3.6.2.1, 3.6.2.2	YES-2, 3	
3/4.7.A.2	Primary Containment Integrity			
3/4.7.A.2.a	Primary Containment Integrity	3.6.1.1, 3.6.1.3, 3.10.1	YES-3	
3/4.7.A.2.b	Deleted by Amendment 132			
3/4.7.A.2.c	Primary Containment Air Lock	3.6.1.1, 3.6.1.2	YES-3	
3/4.7.A.3	Pressure Suppression Chamber - Reactor Building Vacuum Breakers	3.6.1.6	YES-3	
3/4.7.A.4	Pressure Suppression Chamber - Drywell Vacuum Breakers	3.6.1.1, 3.6.1.7	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.

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CURRENT TS (CTS) NUMBER	CURRENT TITLE	NEW TS (ITS) NUMBER	RETAINED/ CRITERION FOR INCLUSION	NOTES(a)
3/4.7.A.5	Primary Containment Oxygen Concentration	3.6.3.1	YES-2	
3/4.7.B	Standby Gas Treatment System	3.6.4.3, 5.5.6	YES-3	
3/4.7.C	Secondary Containment	3.6.4.1, 3.6.4.2	YES-3	
3/4.7.D	Primary Containment Isolation Valves	3.6.1.3, 5.5.11	YES-3	
3/4.7.E	Deleted by Amendment 138			
3/4.8	**Main Condenser Offgas**			
3/4.8.A	Main Condenser Offgas Activity	3.7.6	YES-2	
3/4.9	**Auxiliary Electrical Systems**	**3.8**		
3.9.A	Operational Requirements for Startup	3.8.1, 3.8.4, 3.8.6, 3.8.7	YES-3	
4.9.A	Substation Switchyard Battery	Deleted	NO	Deleted, see technical change discussion in the Discussion of Changes for ITS 3.8.1.
3.9.B	Operational Requirements for Continued Operation	3.8.1, 3.8.3, 3.8.4, 3.8.6, 3.8.7	YES-3	
4.9.B.3	Standby Diesel Generator	3.8.1, 3.8.3, 5.5.8	YES-3	
4.9.B.4	Station Battery Systems	3.8.4, 3.8.6, 3.8.7	YES-3	
4.9.B.5	24V Battery System	Deleted	NO	Deleted, see technical change discussion in the Discussion of Changes for ITS 3.8.4.
3/4.10	**REFUELING**	**3.9**		
3/4.10.A	Refueling Interlocks	3.9.1, 3.9.2	YES-3	
3/4/10.B	Core Monitoring	3.3.1.2	YES	
3/4/10.C	Fuel Storage Pool Water Level	3.7.8	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)
3/4.10.D	Decay Time	Deleted	NO	Deleted, see technical change discussion in the Discussion of Changes for CTS 3.10.D.
3/4.10.E	Extended Core and Control Rod Drive Maintenance	3.10.2, 3.10.6	YES-3	
3/4.11	**REACTOR FUEL ASSEMBLIES**	**3.2**		
3/4.11.A	Average Planar Linear Heat	3.2.1	YES-2	
3/4.11.B	Linear Heat Generation Rate	3.2.3	YES-2	
3/4.11.C	Minimum Critical Power Ratio (MCPR)	3.2.2	YES-2	
3/4.13	**ALTERNATE SHUTDOWN SYSTEM**			
3/4.13.A	Alternate Shutdown System	3.3.3.2	YES-4	
3/4.14	**ACCIDENT MONITORING INSTRUMENTATION**	3.3.3.1, 3.3.6.3	YES-3	See Appendix A, pages 8 and 9. 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-1433, Revision 3.
3/4.17	**CONTROL ROOM HABITABILITY**			
3/4.17.A	Control Room Ventilation System	3.7.5	YES-3	
3/4.17.B	Control Room Emergency Filtration System	3.7.4, 5.5.6	YES-3	
5.0	**DESIGN FEATURES**	**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

3/4.1.A REACTOR PROTECTION SYSTEM

LCO STATEMENT:

The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be given in Table 3.1.1. The time from initiation of any channel trip to the de-energization of the scram pilot valve solenoids shall not exceed 50 milliseconds.

3/4.1.A and B, and Table 3.1.1 Trip Function 9, Table 4.1.1 Instrument Channel 5, and Table 4.1.2 Instrument Channel 7 (Turbine Condenser Low Vacuum).

DISCUSSION:

The turbine condenser low vacuum scram is provided to protect the main condenser from overpressurization in the event that vacuum is lost. A loss of condenser vacuum would cause the turbine stop valves to close, resulting in a turbine trip transient. The low condenser vacuum trip anticipates this transient and scrams the reactor. No design basis accidents or transients take credit for this scram signal.

COMPARISON TO SCREENING CRITERIA:

1. The turbine condenser low vacuum scram instrumentation is not an instrument 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 turbine condenser low vacuum scram instrumentation is not used for, nor capable of, monitoring a process variable that is an initial condition of a DBA or transient analysis.

3. The turbine condenser low vacuum scram instrumentation is not used as part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (item 337) of NEDO-31466, Supplement 1, the loss of the turbine condenser low vacuum scram instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. Nuclear Management Company, LLC has reviewed this evaluation, considers it applicable to Monticello, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the portions of the LCO and Surveillances applicable to the Turbine Condenser Low Vacuum scram instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2.C.1 CONTROL ROD BLOCK ACTUATION

LCO STATEMENT:

The limiting conditions for operation for the instrumentation that actuates control rod block are given in Table 3.2.3.

Table 3.2.3 Function 1, SRM

a. Upscale
b. Detector not fully inserted

DISCUSSION:

SRM signals are used to monitor neutron flux during refueling, shutdown, and startup conditions. When IRMs are not above Range 2, the SRM control rod block functions to prevent a control rod withdrawal if the count rate exceeds a preset value or falls below a preset limit. No design basis accident (DBA) or transient analysis takes credit for rod block signals initiated by the SRMs.

COMPARISON TO SCREENING CRITERIA:

1. The SRM control rod block 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 SRM control rod block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analysis.

3. The SRM control rod block instrumentation is not a part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (item 137) of NEDO-31466, the loss of the SRM control rod block function was found to be a nonsignificant risk contributor to core damage frequency and offsite releases. Nuclear Management Company, LLC has reviewed this evaluation, considers it applicable to Monticello, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Control Rod Block Actuation LCO and Surveillances applicable to SRM instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2.C.1 CONTROL ROD BLOCK ACTUATION

LCO STATEMENT:

The limiting conditions for operation for the instrumentation that actuates control rod block are given in Table 3.2.3.

Table 3.2.3 Function 2, IRM

a. Downscale
b. Upscale

DISCUSSION:

IRMs are provided to monitor the neutron flux levels during refueling, shutdown, and startup conditions. The IRM control rod block functions to prevent a control rod withdrawal if the IRM reading exceeds a preset value, or if the IRM is inoperable. No design basis accident (DBA) or transient analysis takes credit for rod block signals initiated by IRMs.

COMPARISON TO SCREENING CRITERIA:

1. The IRM control rod block 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 IRM control rod block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analysis.

3. The IRM control rod block instrumentation is not a part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (item 138) of NEDO-31466, the loss of the IRM control rod block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. Nuclear Management Company, LLC has reviewed this evaluation, considers it applicable to Monticello, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Control Rod Block Actuation LCO and Surveillances applicable to IRM instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2.C.1 CONTROL ROD BLOCK ACTUATION

LCO STATEMENT:

The limiting conditions for operation for the instrumentation that actuates control rod block are given in Table 3.2.3.

Table 3.2.3 Function 3, APRM

a. Upscale
   (1) TLO Flow Biased
   (2) SLO Flow Biased
   (3) High Flow Clamp

b. Downscale

DISCUSSION:

The APRM control rod block functions to prevent conditions that would require RPS action if allowed to proceed, such as during a "control rod withdrawal error at power." The APRMs utilize LPRM signals to create the APRM rod block signal and provide information about the average core power. However, the rod block function is not used to mitigate a design basis accident (DBA) or transient.

COMPARISON TO SCREENING CRITERIA:

1. The APRM control rod block 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 APRM control rod block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analysis.

3. The APRM control rod block instrumentation is not a part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (item 135) of NEDO-31466, the loss of the APRM control rod block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. Nuclear Management Company, LLC has reviewed this evaluation, considers it applicable to Monticello, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Control Rod Block Actuation LCO and Surveillances applicable to APRM instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2.C.1 CONTROL ROD BLOCK ACTUATION

LCO STATEMENT:

The limiting conditions for operation for the instrumentation that actuates control rod block are given in Table 3.2.3.

Table 3.2.3 Function 5, Scram Discharge Volume

Water Level High
a. East
b. West

DISCUSSION:

The Scram Discharge Volume (SDV) control rod block functions to prevent control rod withdrawals, utilizing SDV signals to create the rod block signal if water is accumulating in the SDV. The purpose of measuring the SDV water level is to ensure that there is sufficient volume remaining to contain the water discharged by the control rod drives during a scram, thus ensuring that the control rods will be able to insert fully. This rod block signal provides an indication to the operator that water is accumulating in the SDV and prevents further rod withdrawals. With continued water accumulation, a reactor protection system initiated scram signal will occur. Thus, the SDV water level rod block signal provides an opportunity for the operator to take action to avoid a subsequent scram. No design basis accident (DBA) or transient takes credit for rod block signals initiated by the SDV instrumentation.

COMPARISON TO SCREENING CRITERIA:

1. The SDV control rod block 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 SDV control rod block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analysis.

3. The SDV control rod block instrumentation is not a part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (item 139) of NEDO-31466, the loss of the SDV control rod block function was found to be a nonsignificant risk contributor to core damage frequency and offsite releases. Nuclear Management Company, LLC has reviewed this evaluation, considers it applicable to Monticello, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the Control Rod Block Actuation LCO and Surveillances applicable to SDV instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

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4.5.A.4 ADS INHIBIT SWITCH

SR STATEMENT:

ADS Inhibit Switch Operability Each Operating Cycle

DISCUSSION

CTS 4.5.A.4 requires the performance of an ADS Inhibit Switch Operability test. The ADS Inhibit Switch allows the operator to defeat ADS actuation as directed by the emergency operating procedures under conditions for which ADS would not be desirable. For example, during an ATWS event low pressure ECCS system activation would dilute sodium pentaborate injected by the Standby Liquid Control (SLC) System thereby reducing the effectiveness of the SLC System ability to shutdown the reactor. While 10 CFR 50.36(c)(2) criteria are not normally used for an individual Surveillance requirement, they are used in this case since the previous BWR Standard Technical Specifications included the ADS Manual Inhibit Switch as a separate Specification and the NRC evaluated it as such as documented in the NRC Staff Review of NSSS Vendor Owners Groups Application of the Commissions Interim Policy Criteria to Standard Technical Specifications, letter dated May 9, 1988. This SR does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual (TRM).

COMPARISON TO THE SCREENING CRITERIA:

1. The ADS Inhibit Switch is not an instrument 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 ADS Inhibit Switch is not used for, nor capable of, monitoring a process variable that is an initial condition of a DBA or transient analysis.

3. The ADS Inhibit Switch is not used as part of a primary success path in the mitigation of a DBA or transient. The inhibit feature was added to allow defeating the automatic ADS function when such action is required by the Emergency Operating Procedures. However, such manual operator action is not credited in a design basis accident or transient analysis.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (item 112B) of NEDO-31466, the loss of the ADS Inhibit switch was found to be a non-significant risk contributor to core damage frequency and offsite releases. Nuclear Management Company, LLC has reviewed this evaluation, considers it applicable to Monticello, and concurs with the assessment.

CONCLUSION:

Since the screening criteria have not been satisfied, the portions of the LCO and Surveillances applicable to the ADS Manual Inhibit switch may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.6.C.2 and 3/4.6.C.3 REACTOR COOLANT WATER CHEMISTRY

LCO STATEMENT:

3/4.6.C.2. (a) The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour except as specified in 3.6.C.2.b.

Conductivity	5 μmho/cm
Chloride ion	0.1 ppm

3/4.6.C.2. (b) For reactor startups the maximum value for conductivity shall not exceed μmho/cm and the maximum value for chloride ion concentration shall not exceed 0. 1 ppm for the first 24 hours after placing the reactor in the power operating condition.

3/4.6.C.3.) Except as specified in 3. 6.C. 2.b above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100, 000 lbs. per hour.

Conductivity	5 μmho/cm
Chloride ion	0.5 ppm

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 Sections 3.5 and 6, and summarized in Table 4-1 (item 211) of NEDO-31466, the reactor coolant water chemistry was found to be a non-significant risk contributor to core damage frequency and offsite releases. Nuclear Management Company, LLC has reviewed this evaluation, considers it applicable to Monticello, 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.

3/4.14.F ACCIDENT MONITORING

LCO STATEMENT:

Whenever irradiated fuel is in the reactor vessel and reactor coolant water temperature is greater than 212°F, the limiting conditions for operation for accident monitoring instrumentation given in Table 3.14.1 shall be satisfied.

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.

COMPARISON TO SCREENING CRITERIA:

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 Monticello 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.

The following summarizes the Nuclear Management Company, LLC position for those instruments currently in Monticello Technical Specifications.

Type A Variables

1. Reactor Vessel Fuel Zone Water Level
2. Suppression Pool Temperature

Other Type, Category 1 Variables

1. Drywell Wide Range Pressure
2. Suppression Pool Wide Range Level
3. Drywell High Range Radiation

For other post-accident monitoring instrumentation currently in Technical Specifications, their loss is not risk-significant since the variables they monitor did not qualify as a Type A or Category 1 variable (one that is important to safety and needed by the operator, so that the operator can perform necessary normal actions).

CONCLUSION:

Since the screening criteria have not satisfied for non-Regulatory Guide 1.97 Type A or Category 1 variable instruments, their associated LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications. The instruments to be relocated are as follows:

1. Safety/Relief Valve Position
2. Offgas Stack Wide Range Radiation
3. Reactor Bldg Vent Wide Range Radiation

Summary of Changes
Generic Determination of No Significant Hazards Consideration
and Environmental Assessment

No changes are required for the Volume.

ATTACHMENT 1

VOLUME 2

MONTICELLO
IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

GENERIC DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION AND ENVIRONMENTAL ASSESSMENT

Revision 1

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
ADMINISTRATIVE CHANGES

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." Some of the proposed changes involve reformatting, renumbering, and rewording of Current Technical Specifications (CTS) with no change in intent. These changes, since they do not involve technical changes to the CTS, are administrative.

This type of change is connected with the movement of requirements within the current requirements, or with the modification of wording that does not affect the technical content of the CTS. These changes also include non-technical modifications of requirements to conform to NEI 01-03, "Writer's Guide for the Improved Standard Technical Specifications," or provide consistency with the Improved Standard Technical Specifications in NUREG-1433. Administrative changes are not intended to add, delete, or relocate any technical requirements of the CTS.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change involves reformatting, renumbering, and rewording the CTS. The reformatting, renumbering, and rewording process involves no technical changes to the CTS. As such, this change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. This change is administrative in nature. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
MORE RESTRICTIVE CHANGES

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." Some of the proposed changes involve adding more restrictive requirements to the Current Technical Specifications (CTS) by either making current requirements more stringent or by adding new requirements that currently do not exist.

These changes include additional requirements that decrease allowed outage times, increase the Frequency of Surveillances, impose additional Surveillances, increase the scope of Specifications to include additional plant equipment, increase the Applicability of Specifications, or provide additional actions. These changes are generally made to conform with NUREG-1433 and have been evaluated to not be detrimental to plant safety.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change provides more stringent Technical Specification requirements for the facility. These more stringent requirements do not result in operations that significantly increase the probability of initiating an analyzed event, and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change does impose different Technical Specification requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The imposition of more restrictive requirements either has no effect on or increases the margin of plant safety. As provided in the discussion of change, each change in this category is, by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
RELOCATED SPECIFICATIONS

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." Some of the proposed changes involve relocating Current Technical Specification (CTS) Limiting Conditions for Operations (LCOs) to licensee controlled documents.

NMC has evaluated the CTS using the criteria set forth in 10 CFR 50.36. Specifications identified by this evaluation that did not meet the retention requirements specified in the regulation are not included in the ITS. These specifications have been relocated from the CTS to the Technical Requirements Manual, which is incorporated into the Updated Safety Analysis Report (USAR).

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relocates requirements and Surveillances for structures, systems, components, or variables that do not meet the criteria of 10 CFR 50.36 (c)(2)(ii) for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the Monticello Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and Surveillances for these affected structures, systems, components, or variables will be relocated from the CTS to an appropriate administratively controlled document which will be incorporated into the USAR, thus it will be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components, or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59, and are subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose or

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

eliminate any requirements, and adequate control of existing requirements will be maintained. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change will not reduce a margin of safety because it has no significant effect on any safety analyses assumptions, as indicated by the fact that the requirements do not meet the 10 CFR 50.36 criteria for retention. In addition, the relocated requirements are moved without change, and any future changes to these requirements will be evaluated per 10 CFR 50.59.

   NRC prior review and approval of changes to these relocated requirements, in accordance with 10 CFR 50.92, will no longer be required. This review and approval does not provide a specific margin of safety that can be evaluated. However, the proposed change is consistent with NUREG-1433, issued by the NRC, which allows revising the CTS to relocate these requirements and Surveillances to a licensee controlled document controlled by 10 CFR 50.59. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
REMOVED DETAIL CHANGES

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." Some of the proposed changes involve moving details out of the Current Technical Specifications (CTS) and into the Technical Specifications Bases, the Updated Safety Analysis Report (USAR), the Technical Requirements Manual (TRM), or other documents under regulatory control such as the CORE OPERATING LIMITS REPORT (COLR), Offsite Dose Calculation Manual (ODCM), Operational Quality Assurance Program (OQAP), Inservice Testing Program (IST), and Inservice Inspection Program (IIP). The removal of this information is considered to be less restrictive because it is no longer controlled by the Technical Specification change process. Typically, the information moved is descriptive in nature and its removal conforms with NUREG-1433 for format and content.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relocates certain details from the CTS to other documents under regulatory control. The Technical Specification Bases and TRM will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the ITS. The USAR is subject to the change control provisions of 10 CFR 50.59 or 10 CFR 50.71(e). Other documents are subject to controls imposed by ITS or regulations. Since any changes to these documents will be evaluated, no significant increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operations. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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**DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES**

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change will not reduce a margin of safety because it has no effect on any assumption of the safety analyses. In addition, the details to be moved from the CTS to other documents are not being changed. Since any future changes to these details will be evaluated under the applicable regulatory change control mechanism, no significant reduction in a margin of safety will be allowed. A significant reduction in the margin of safety is not associated with the elimination of the 10 CFR 50.90 requirement for NRC review and approval of future changes to the relocated details. Not including these details in the Technical Specifications is consistent with NUREG-1433, issued by the NRC, which allows revising the Technical Specifications to relocate these requirements and Surveillances to a licensee controlled document controlled by 10 CFR 50.59, 10 CFR 50.71(e), or other Technical Specification controlled or regulation controlled documents. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 1
RELAXATION OF LCO REQUIREMENT

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4" (ISTS). Some of the proposed changes involve relaxation of the Current Technical Specification (CTS) Limiting Conditions for Operation (LCOs) by the elimination of specific items from the LCO or Tables referenced in the LCO, or the addition of exceptions to the LCO.

These changes reflect the ISTS approach to provide LCO requirements that specify the protective conditions that are required to meet safety analysis assumptions for required features. These conditions replace the lists of specific devices used in the CTS to describe the requirements needed to meet the safety analysis assumptions. The ITS also includes LCO Notes which allow exceptions to the LCO for the performance of testing or other operational needs. The ITS provides the protection required by the safety analysis, and provides flexibility for meeting the conditions without adversely affecting operations since equivalent features are required to be OPERABLE. The ITS is also consistent with the plant current licensing basis, as may be modified in the discussion of individual changes. These changes are generally made to conform with NUREG-1433, and have been evaluated to not be detrimental to plant safety.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change provides less restrictive LCO requirements for operation of the facility. These less restrictive LCO requirements do not result in operation that will significantly increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event in that the requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the current safety analyses and licensing basis. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does impose different

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requirements. However, the change is consistent with the assumptions in the current safety analyses and licensing basis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

The imposition of less restrictive LCO requirements does not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to ensure that the current safety analyses and licensing basis requirements are maintained. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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**DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES**

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 2
RELAXATION OF APPLICABILITY

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." Some of the proposed changes involve relaxation of the applicability of Current Technical Specification (CTS) Limiting Conditions for Operation (LCOs) by reducing the conditions under which the LCO requirements must be met.

Reactor operating conditions are used in CTS to define when the LCO features are required to be OPERABLE. CTS Applicabilities can be specific defined terms of reactor conditions or more general such as "the reactor shall not be made critical." Generalized applicability conditions are not contained in ITS, therefore the ITS eliminates CTS requirements such as "the reactor shall not be made critical" replacing them with ITS defined MODES or applicable conditions that are consistent with the application of the plant safety analyses assumptions for OPERABILITY of the required features.

CTS requirements may also be eliminated during conditions for which the safety function of the specified safety system is met because the feature is performing its intended safety function. Deleting applicability requirements that are indeterminate or which are inconsistent with application of accident analyses assumptions is acceptable because when LCOs cannot be met, the ITS may be satisfied by exiting the applicability which takes the plant out of the conditions that require the safety system to be OPERABLE.

This change provides the protection required by the safety analyses, and provides flexibility for meeting limits by restricting the application of the limits to the conditions assumed in the safety analyses. The ITS is also consistent with the plant current licensing basis, as may be modified in the discussion of individual changes. The change is generally made to conform with NUREG-1433, and has been evaluated to not be detrimental to plant safety.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes the conditions under which the LCO requirements for operation of the facility must be met. These less restrictive applicability requirements for the LCOs do not result in operation that will significantly increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event in that the requirements continue to ensure that process variables, structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does impose different requirements. However, the requirements are consistent with the assumptions in the safety analyses and licensing basis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The relaxed applicability of LCO requirements does not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to ensure that the LCO requirements are applied in the MODES and specified conditions assumed in the safety analyses and licensing basis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 3
RELAXATION OF COMPLETION TIME

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." Some of the proposed changes involve relaxation of the Completion Times for Required Actions in the Current Technical Specifications (CTS).

Upon discovery of a failure to meet a Limiting Condition for Operation (LCO), the ITS specifies times for completing Required Actions of the associated ITS Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken within specified Completion Times. These times define limits during which operation in a degraded condition is permitted. Adopting Completion Times from the ITS is acceptable because the Completion Times take into account the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a Design Basis Accident (DBA) occurring during the repair period. In addition, the ITS provides consistent Completion Times for similar conditions. These changes are generally made to conform with NUREG-1433, and have been evaluated to not be detrimental to plant safety.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes the Completion Time for a Required Action. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated, and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Completion Time does not significantly increase the probability of any accident previously evaluated. The consequences of an analyzed accident during the relaxed Completion Time are the same as the consequences during the existing Completion Time. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

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GENERIC CHANGES

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the method governing normal plant operation. The Required Actions and associated Completion Times in the ITS have been evaluated to ensure that no new accident initiators are introduced. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

The relaxed Completion Time for a Required Action does not involve a significant reduction in the margin of safety. As provided in the discussion of change, the change has been evaluated to ensure that the allowed Completion Time is consistent with safe operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 4
RELAXATION OF REQUIRED ACTION

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." Some of the proposed changes involve relaxation of the Required Actions in the Current Technical Specifications (CTS).

Upon discovery of a failure to meet a Limiting Condition for Operation (LCO), the ITS specifies Required Actions to complete for the associated Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken in response to the degraded conditions. These actions minimize the risk associated with continued operation while providing time to repair inoperable features. Some of the Required Actions are modified to place the plant in a MODE in which the LCO does not apply. Adopting Required Actions from NUREG-1433 is acceptable because the Required Actions take into account the OPERABILITY status of redundant systems of required features, the capacity and capability of the remaining features, and the compensatory attributes of the Required Actions as compared to the LCO requirements. These changes are generally made to conform with NUREG-1433, and have been evaluated to not be detrimental to plant safety.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes Required Actions. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated, and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Required Actions do not significantly increase the probability of any accident previously evaluated. The Required Actions in the ITS have been developed to provide appropriate remedial actions to be taken in response to the degraded condition considering the OPERABILITY status of the redundant systems of required features, and the capacity and capability of remaining features while minimizing the risk associated with continued operation. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The Required Actions and associated Completion Times in the ITS have been evaluated to ensure that no new accident initiators are introduced. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The relaxed Required Actions do not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to minimize the risk of continued operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a Design Basis Accident (DBA) occurring during the repair period. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 5
DELETION OF SURVEILLANCE REQUIREMENT

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." Some of the proposed changes involve deletion of Surveillance Requirements in the Current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified OPERABLE prior to entering applicable operating conditions. The ITS eliminates unnecessary CTS Surveillance Requirements that do not contribute to verification that the equipment used to meet the Limiting Condition for Operation (LCO) can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. These changes are generally made to conform with NUREG-1433, and have been evaluated to not be detrimental to plant safety.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change deletes Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be OPERABLE and capable of performing the accident mitigation functions assumed in the accident analyses. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The remaining Surveillance Requirements are consistent with industry practice, and are considered to be sufficient to prevent the removal of the subject Surveillances from creating a new or different type of accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The deleted Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the change has been evaluated to ensure that the deleted Surveillance Requirements are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 6
RELAXATION OF SURVEILLANCE REQUIREMENT ACCEPTANCE CRITERIA

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." Some of the proposed changes involve the relaxation of Surveillance Requirements acceptance criteria in the Current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified OPERABLE prior to entering applicable operating conditions. The ITS eliminates or relaxes the Surveillance Requirement acceptance criteria that do not contribute to verification that the equipment used to meet the Limiting Condition for Operation (LCO) can perform its required functions. For example, the ITS allows some Surveillance Requirements to verify OPERABILITY under actual or test conditions. Adopting the ITS allowance for "actual" conditions is acceptable because required features cannot distinguish between an "actual" signal or a "test" signal. Also included are changes to CTS requirements that are replaced in the ITS with separate and distinct testing requirements that when combined, include OPERABILITY verification of all components required in the LCO for the features specified in the CTS. Adopting this format preference in the ITS is acceptable because Surveillance Requirements that remain include testing of all previous features required to be verified OPERABLE. Changes that provide exceptions to Surveillance Requirements to provide for variations that do not affect the results of the test are also included in this category. These changes are generally made to conform with NUREG-1433, and have been evaluated to not be detrimental to plant safety.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes the acceptance criteria of Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be OPERABLE and capable of performing the accident mitigation functions assumed in the accident analyses. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The relaxed acceptance criteria for Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the relaxed Surveillance Requirement acceptance criteria have been evaluated to ensure that they are sufficient to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner that gives confidence that the equipment can perform its assumed safety function. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 7
RELAXATION OF SURVEILLANCE FREQUENCY, NON-24 MONTH TYPE CHANGE

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." Some of the proposed changes involve the relaxation of Surveillance Frequencies in the Current Technical Specifications (CTS).

CTS and ITS Surveillance Frequencies specify time interval requirements for performing Surveillance tests. Increasing the time interval between Surveillance tests in the ITS results in decreased equipment unavailability due to testing which also increases equipment availability. In general, the ITS contain Surveillance Frequencies that are consistent with industry practice or industry standards for achieving acceptable levels of equipment reliability. Adopting testing practices specified in the ITS is acceptable based on similar design, like-component testing for the system application and the availability of other ITS requirements which provide regular checks to ensure limits are met. Relaxation of Surveillance Frequency can also include the addition of Surveillance Notes which allow testing to be delayed until appropriate unit conditions for the test are established, or exempt testing in certain MODES or specified conditions in which the testing can not be performed.

Reduced testing can result in a safety enhancement because the unavailability due to testing is reduced, and reliability of the affected structure, system or component should remain constant or increase. Reduced testing is acceptable where operating experience, industry practice, or the industry standards such as manufacturers' recommendations have shown that these components usually pass the Surveillance when performed at the specified interval, thus the Surveillance Frequency is acceptable from a reliability standpoint. Surveillance Frequency changes to incorporate alternate train testing have been shown to be acceptable where other qualitative or quantitative test requirements are required that are established predictors of system performance. Surveillance Frequency extensions can be based on NRC-approved topical reports. The NRC staff has accepted topical report analyses that bound the plant-specific design and component reliability assumptions. These changes are generally made to conform with NUREG-1433, and have been evaluated to not be detrimental to plant safety.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change relaxes Surveillance Frequencies. The relaxed Surveillance Frequencies have been established based on achieving acceptable levels of equipment reliability. Consequently, equipment that could initiate an accident previously evaluated will continue to operate as expected, and the probability of the initiation of any accident previously evaluated will not be significantly increased. The equipment being tested is still required to be OPERABLE and capable of performing any accident mitigation functions assumed in the accident analyses. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The relaxed Surveillance Frequencies do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the relaxation in the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Thus, appropriate equipment continues to be tested at a Frequency that gives confidence that the equipment can perform its assumed safety function when required. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 8
DELETION OF REPORTING REQUIREMENT

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." Some of the proposed changes involve the deletion of requirements in the Current Technical Specifications (CTS) to send reports to the NRC.

The CTS includes requirements to submit reports to the NRC under certain circumstances. However, the ITS eliminates these requirements for many such reports and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The ITS changes to reporting requirements are acceptable because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, this change has no effect on the safe operation of the plant. These changes are generally made to conform with NUREG-1433, and have been evaluated to not be detrimental to plant safety.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change deletes reporting requirements. Sending reports to the NRC is not an initiator of any accident previously evaluated. Consequently, the probability of any accident previously evaluated is not significantly increased. Sending reports to the NRC has no effect on the ability of equipment to mitigate an accident previously evaluated. As a result, the consequences of any accident previously evaluated is not significantly affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The deletion of reporting requirements does not result in a significant reduction in the margin of safety. The ITS eliminates the requirements for many such reports and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The change to reporting requirements does not affect the margin of safety because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 9
DELETION OF SURVEILLANCE REQUIREMENT SHUTDOWN PERFORMANCE REQUIREMENTS

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." Some of the proposed changes involve the deletion of the requirement to perform Surveillance Requirements while in a shutdown condition in the Current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified OPERABLE periodically. The CTS requires these Surveillances to be performed with the unit in a specified condition, usually in a condition outside the Applicability of the Limiting Condition for Operation (LCO). The ITS Surveillance does not include the restriction on unit conditions. The control of the unit conditions appropriate to perform the test is an issue for procedures and scheduling, and has been determined by the NRC Staff to be unnecessary as an ITS restriction. As indicated in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991, allowing this control is consistent with the vast majority of other Technical Specification Surveillances that do no dictate unit conditions for the Surveillance. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence that the equipment can perform its assumed safety function. These changes are made to conform with NUREG-1433 and have been evaluated to not be detrimental to plant safety.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change involves the deletion of the requirement to perform Surveillance Requirements while in a shutdown condition. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The appropriate plant conditions for performance of the Surveillance will continue to be controlled in plant procedures to assure the potential consequences are not significantly increased. This control method has been previously determined to be acceptable as indicated in NRC Generic Letter No. 91-04. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident because of the availability of redundant systems or equipment. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC CHANGES

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change involves the deletion of the requirement to perform Surveillance Requirements while in a shutdown condition, but does not change the method of performance. The appropriate plant conditions for performance of the Surveillance will continue to be controlled in plant procedures to assure the possibility of a new or different kind of accident are not created. The control method has been previously determined to be acceptable as indicated in NRC Generic Letter No. 91-04. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change involves the deletion of the requirement to perform Surveillance Requirements while in a shutdown condition. However, the appropriate plant conditions for performance of the Surveillance will continue to be controlled in plant procedures. The control method has been previously determined to be acceptable as indicated in NRC Generic Letter No. 91-04. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS GENERIC CHANGES

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 10
CHANGING INSTRUMENTATION ALLOWABLE VALUES

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." Some of the proposed changes to the Current Technical Specifications (CTS) involve a change to the Allowable Values for Technical Specification instrumentation.

The proposed changes in selected Allowable Values for the instrumentation included in Section 3.3 of the ITS have been established using the GE setpoint methodology guidance, as specified in the Monticello setpoint methodology. The analytic limits are derived from limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits. The difference between the analytic limit and the Allowable Value allows for channel instrument accuracy, calibration accuracy, process measurement accuracy, and primary element accuracy. The margin between the Allowable Value and the nominal trip setpoint (NTSP) allows for instrument drift that might occur during the established surveillance period. Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift. These two verifications are statistical evaluations to provide additional assurance of the acceptability of the NTSP and may require changes to the NTSP. Use of these methods and verifications provides the assurance that if the setpoint is found conservative to the Allowable Value during surveillance testing, the instrumentation would have provided the required trip function by the time the process reached the analytic limit for the applicable events.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change involves the change in selected Allowable Values for the instrumentation included in Section 3.3 of the ITS. The proposed changes will not result in any hardware changes. The instrumentation included in the proposed Section 3.3 of the ITS is not assumed to be an initiator of any analyzed event. Existing operating margin between plant conditions and actual plant setpoints is not significantly reduced due to this proposed change. As a result, the proposed change will not result in unnecessary plant transients. The role of the instrumentation included in Section 3.3 of the ITS is in mitigating and thereby limiting the consequences of accidents. The Allowable Values have been developed to ensure that the design and safety analyses limits will be satisfied. The methodology used for the development of the Allowable Values ensures that the affected instrumentation remains capable of mitigating design basis events as described in the safety analyses, and that the results and consequences described in the safety analyses remain bounding. Additionally, the proposed change does not alter the ability of the instrumentation and associated systems and components to detect and mitigate events. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed changes have been established using the GE setpoint methodology guidance, as specified in the Monticello setpoint methodology, and do not create the possibility of a new or different kind of accident from any accident previously evaluated. This is based on the fact that the method and manner of plant operation is unchanged. The use of the proposed Allowable Values does not impact safe operation of the plant, in that the safety analyses limits will be maintained. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). These Allowable Values were developed using a methodology to ensure the affected instrumentation and associated systems and components remain capable of mitigating accidents and transients. Plant equipment will not be operated in a manner different from previous operation, except that setpoint may be changed. Since operational methods remain unchanged, and the existing operating parameters have been evaluated to maintain the unit within existing design basis criteria, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change does not involve a reduction in a margin of safety. The proposed changes have been developed using a methodology to ensure safety analyses limits are not exceeded. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

ENVIRONMENTAL ASSESSMENT

Nuclear Management Company, LLC (NMC) has evaluated this license amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. NMC has determined that this license amendment meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in the generic and specific Determination of No Significant Hazards Considerations, this proposed amendment does not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed amendment does not affect the generation of any radioactive effluents, and does not affect any of the permitted effluent release paths.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

No new effluents or effluent release paths are created by the proposed amendment.

Therefore, pursuant to 10 CFR 51.22 (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

Summary of Changes
ITS Chapter 1.0

Change Description	Affected Pages
The changes described in the NMC response to Question 200510141334 (in Section 3.1) have been made. Changes are made to be consistent with TSTF-439, Rev. 2 (Eliminate Second Completion Times Limiting Time From Discovery of Failure to Meet an LCO).	Pages 45, 48, 49, 50, and 63 of 71
The changes described in the NMC response to Question 200512151125 have been made. Changes are made to be consistent with TSTF-485, Rev. 0 (Correct Example 1.4-1).	Pages 58 and 63 of 71
The changes described in the NMC response to Question 200601201446 have been made. Minor editorial changes are made.	Page 56 of 71

ATTACHMENT 1

VOLUME 3

MONTICELLO
IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 1.0
USE AND APPLICATION

Revision 1

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS Chapter 1.0, Use and Application

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

INTRODUCTION

~~These Technical Specifications are prepared in accordance with the requirements of 10 CFR 50.36 and apply to the Monticello Nuclear Generating Plant, Unit No. 1. The bases for these Specifications are included for information and understandability purposes.~~

1.1 1.0 DEFINITIONS

A.2 INSERT 1

~~The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the Specifications may be achieved.~~ INSERT 2 L.1

CORE ALTERATION shall be the movement

CORE A. ~~Alteration of the Reactor Core~~ - ~~The act of moving~~ any component in the region ~~above the core support plate, below the upper grid and within the shroud~~ with the vessel head removed and fuel in the reactor vessel. (Normal operating functions such as control rod movement using the normal drive mechanism, tip scans, SRM and IRM detector movements, etc., are not to be considered core alterations.) within the reactor vessel M.1 INSERT 3 L.1 INSERT 4 A.3 M.2

Table 1.1-1 MODE 2 ~~B. Hot Standby - Hot Standby means operation with the reactor critical in the startup mode at a power level just sufficient to maintain reactor pressure and temperature.~~ Add proposed Table 1.1-1 MODE 2 M.3

C. (Deleted)

1.3 ~~D.~~ Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action. A.5

1.1 CHANNEL E. ~~Instrument~~ Functional Test - An ~~instrument~~ functional test ~~means~~ shall be the injection of a simulated or actual signal into the ~~primary~~ channel sensor as close to the sensor as practicable to verify ~~proper instrument channel response, alarm, and/or initiating action~~. L.2 L.3 INSERT 5 INSERT 6 A.6

A.2 **INSERT 1**

-- NOTE --
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

L.1 **INSERT 2**

of any fuel, sources, or reactivity control components,

INSERT 3

The following exceptions are not considered to be CORE ALTERATIONS: M.2 L.1

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and L.1

b. Control rod movement, provided there are no fuel assemblies in the associated core cell. M.2 L.1

A.3 **INSERT 4**

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

A.6 **INSERT 5**

OPERABILITY of all devices in the channel required for channel OPERABILITY

A.6 **INSERT 6**

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.

ITS

1.1 F. ~~Instrument Calibration~~ - ~~An instrument calibration means~~ the adjustment of ~~an instrument signal~~ output ~~so~~ that it ~~corresponds~~ within ~~acceptable~~ range, accuracy ~~and response time~~ to ~~a~~ known value~~(s)~~ of the parameter ~~which~~ the ~~instrument monitors.~~ ~~Calibration~~ shall encompass ~~the entire instrument including actuation, alarm or trip.~~ ~~Response time is not part of the routine instrument calibration but will be checked once per cycle.~~

CHANNEL | and | shall be | , as necessary, | the channel | such | responds | the necessary | that | channel | The CHANNEL | INSERT 7 | INSERT 8

A.7

A.7

~~G. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.~~

A.8

H. ~~-Deleted-~~

~~I. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation, the safety limits will never be exceeded.~~

A.8

J. Minimum Critical Power Ratio (MCPR) - The ~~minimum critical power ratio~~ ~~is the value of~~ critical power ratio ~~associated with the most limiting assembly~~ in the ~~reactor~~ core. ~~Critical power ratio~~ (CPR) is ~~the ratio of~~ that power in ~~a fuel~~ assembly ~~which~~ is calculated by ~~the GEXL correlation~~ to cause some point in the assembly to experience boiling transition ~~to~~ the actual assembly operating power.

MCPR | shall be | the smallest | (CPR) | that exists | the | that | The | , divided by | for each class of fuel | application of the appropriate correlation(s)

A.9

K. Mode - ~~The reactor mode is that which is established by the mode-selector switch.~~

INSERT 9

M.3

L. Operable - A system, subsystem, ~~train~~, component or device shall be ~~Operable~~ or have ~~Operability~~ when it is capable of performing its specified function(s). ~~Implicit in this definition shall be the assumption that~~ all necessary attendant instrumentation, controls, normal ~~and~~ emergency electrical power ~~sources~~, cooling ~~or~~ seal water, lubrication ~~or~~ other auxiliary equipment that are required for the system, subsystem, ~~train~~, component, or device to perform its function(s) are also capable of performing their related support function(s).

- OPERABILITY | division | , | safety | and when | , and | or | division | , | and | specified safety

A.10

A.11

A.10

Attachment 1, Volume 3, Rev. 1, Page 7 of 71

A.7 **INSERT 7**

all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST

A.7 **INSERT 8**

Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.

M.3 **INSERT 9**

A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

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When a system, subsystem, train, component or device is determined to be inoperable soley because its emergency power source is inoperable, or soley because its normal power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation provided: (1) its corresponding normal or emergency power source is operable; and (2) all of its redundant system(s), subsystem(s), trains(s), component(s) and device(s) are Operable, or likewise satisfy the requirements of this paragraph.

See ITS 3.8.1

A.8

M. Operating - Operating means that a system or component is performing its specified functions.

N. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

Add proposed Table 1.1-1 MODES 1 and 2

Table 1.1-1 MODES 1 and 2

O. Power Operation - Power Operation is any operation with the mode switch in the "Start-Up" or "Run" position with the reactor critical and above 1% rated thermal power.

M.3

P. Primary Containment Integrity - Primary Containment Integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

See ITS 3.6.1.1

A.8

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

See ITS 3.6.1.3

2. At least one door in the airlock is closed and sealed.

See ITS 3.6.1.2

3. All automatic containment isolation valves are operable or are deactivated in the closed position or at least one valve in each line having an inoperable valve is closed

See ITS 3.6.1.3

4. All blind flanges and manways are closed.

See ITS 3.6.1.1 and ITS 3.6.1.3

Q. Protective Instrumentation Logic Definitions

1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system, a single trip signal related to the plant parameter monitored by that instrument channel.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate a protection action. A trip system may require one or more instrument channel trip signals related to one or more plant parameters to initiate trip system action. Initiation of the protective function may require tripping of a single trip system (e.g., HPCI system isolation, off-gas system isolation, reactor building isolation and standby gas treatment initiation, and rod block), or the coincident tripping of two trip systems (e.g., initiation of scram, reactor isolation, and primary containment isolation).

3. Protective Action - An action initiated by the protection system when a limit is exceeded. A protective action can be at channel or system level.

A.8

Attachment 1, Volume 3, Rev. 1, Page 9 of 71

A.1 A.2

ITS

4. Protective Function - A ~~system protective~~ action which results from the protective action of ~~the channels monitoring a particular plant condition.~~ A.8

R. ~~Rated Neutron Flux - Rated flux is the neutron flux that corresponds to a steady-state power level of 1775 thermal megawatts.~~

1.1 S. Rated Thermal Power (RTP) - ~~Rated thermal power means a steady-state power level~~ of 1775 ~~thermal megawatts~~ MWt

RTP shall be a total reactor core heat transfer rate to the reactor coolant

T. ~~Reactor Coolant System Pressure or Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those existing in the vessel steam space.~~ A.8 A.17

U. Refueling Operation and Refueling ~~Outage~~ Interval - ~~Refueling Operation is any operation when the reactor water temperature is less than 212°F and movement of fuel or core components is in progress.~~ For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage. A.17

Refueling Interval is a designated frequency for performing surveillances of once per 24 months.

V. ~~Safety Limit - The safety limits are limits below which the maintenance of the cladding and primary system integrity are assured. Exceeding such a limit is cause for plant shutdown and review by the Commission before resumption of plant operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.~~ A.8

W. Secondary Containment Integrity - Secondary Containment Integrity means that the reactor building is closed ~~and the following conditions are met:~~ See ITS 3.6.4.1 and ITS 3.6.4.2

1. At least one door in each access opening is closed. See ITS 3.6.4.1

2. The standby gas treatment system is operable. See ITS 3.6.4.3

3. All reactor building ventilation system automatic isolation valves are operable or are secured in the closed position. See ITS 3.6.4.2

CHANNEL CHECK shall be the

X. ~~Sensor~~ CHANNEL ~~Check~~ CHECK - A qualitative ~~determination~~ assessment, ~~of operability~~ by observation of , channel ~~sensor~~ behavior during operation. This determination shall include, where possible, comparison ~~with~~ of the channel indication and status to other ~~independent sensors~~ indications or status derived from independent instrument channels measuring the same ~~variable~~ parameter.

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See ITS 3.3.2.1

M.3

See ITS 3.3.2.1

Y. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. In this condition, a reactor scram is initiated and a rod block is inserted directly from the mode switch. The scram can be reset after a short time delay.

LA.1

Add proposed Table 1.1-1 footnote (a)

Table 1.1-1 MODE 3

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

Table 1.1-1 MODE 4

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

M.3

Add Table 1.1-1 MODE 5 and footnote(b)

Z. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

A.8

AA. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling, also referred to as partial nucleate boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

1.1 LEAKAGE — LEAKAGE shall be:

d AB. Pressure Boundary Leakage - ~~Pressure boundary leakage shall be~~ leakage through a nonisolable fault in the (a) reactor coolant system (RCS) ~~pressure boundary~~ component body, pipe wall, or vessel wall.

A.13

a AC. Identified Leakage ~~- Identified leakage shall be:~~

1. Leakage into the drywell, such as that from pump seals or valve packing leaks, that is captured and conducted to a sump or collecting tank; or

2. Leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be Pressure Boundary Leakage;

b AD. Unidentified Leakage - All leakage into the drywell that is not identified Leakage;

c AE. Total Leakage - Sum of the identified and unidentified Leakage; and

A.12

~~AF. through AH. (Deleted)~~

AI. Purging - Purging is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

AJ. Venting - Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required.

A.8

ITS

A.1 A.2

shall be that

that

1.1 AK. ~~Dose Equivalent I-131~~ - Dose Equivalent I-131 is the concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or in NRC Regulatory Guide 1.109, Rev 1, October, 1977.

AEC, 1962,

those listed in Table E-7 of

NRC

~~AL. through AP. (Deleted)~~

(COLR)

COLR

cycle specific parameter

AQ. Core Operating Limits Report - The Core Operating Limits Report is the unit specific document that provides core operating limits for the current operating reload cycle. These cycle specific operating limits shall be determined for each reload cycle in accordance with Specification ~~6.7.A.7~~. Plant operation within these operating limits is addressed in individual specifications.

5.6.3

~~AR. Allowable Value - The Allowable Value is the limiting value of the sensed process variable at which the trip setpoint may be found during instrument surveillance.~~ A.8

INSERT 10 A.14

INSERT 11 M.3

Add proposed ITS Sections
1.2 – Logical Connectors
1.3 – Completion Times
1.4 - Frequency

A.15

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A.14

INSERT 10

ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	The APLHGR shall be applicable to a specific planar height and is equal to the sum of the LHGRs for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at the height.
LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME shall be that time interval from when the main turbine trip solenoid is activated until 80% of the turbine bypass capacity is established. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

M.3

INSERT 11

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel[a] or Startup/Hot Standby	NA
3	Hot Shutdown[a]	Shutdown	> 212
4	Cold Shutdown[a]	Shutdown	≤ 212
5	Refueling[b]	Shutdown or Refuel	NA

M.3

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

A.1 A.2

ITS

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3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3.1 REACTOR PROTECTION SYSTEM	**4.1 REACTOR PROTECTION SYSTEM**
Applicability: Applies to the instrumentation and associated devices which initiate a reactor scram.	Applicability: Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.
Objective: To assure the operability of the reactor protection system.	Objective: To specify the type and frequency of surveillance to be applied to the instrumentation that initiates a scram to verify its operability.
Specification: A. The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1. The time from initiation of any RPS channel trip to the de-energization of the scram pilot valve solenoids ~~shall not exceed 50 milliseconds~~. INSERT 12	Specification: A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively.

See ITS 3.3.1.1

1.1 REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

The RPS RESPONSE TIME shall be that time interval

A.16

A.1

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A.16 **INSERT 12**

The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3.3 CONTROL ROD SYSTEM	4.3 CONTROL ROD SYSTEM
Applicability:	Applicability:
Applies to the operational status of the control rod system.	Applies to the surveillance requirements of the control rod system.
Objective:	Objective:
To assure the ability of the control rod system to control reactivity.	To verify the ability of the control rod system to control reactivity.
Specification:	Specification:
A. Reactivity Limitations	A. Reactivity Limitations
1. Reactivity margin - core loading	1. Reactivity margin - core loading
The core loading shall be limited to that which can be made subcritical in the most reactive condition during the operating cycle with the strongest operable control rod in its full-out position and all other operable rods fully inserted.	Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.25 per cent Δk that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.

1.1 SHUTDOWN MARGIN (SDM)

SDM shall be the amount of reactivity by

the reactor is

INSERT 14

INSERT 13

See ITS 3.1.1

A.4

See ITS 3.1.1

A.4 **INSERT 13**

or would be subcritical assuming that:

a. The reactor is xenon free;

b. The moderator temperature is 68°F; and

A.4 **INSERT 14**

c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

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DISCUSSION OF CHANGES
ITS CHAPTER 1.0, USE AND APPLICATION

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 The CTS Section 1.0 Definition introduction states "The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the Specifications may be achieved." The Note to ITS Section 1.1 states "The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases." This changes the CTS by replacing the CTS Section 1.0 introduction of the definitions with a Note.

The ITS Section 1.0 Note serves the same purpose of the CTS Section 1.0 introduction. A major change to the ITS definitions is that the entire defined term is capitalized in ITS Section 1.1 instead of just the first letter in the CTS. In addition, whenever the term is used throughout the Technical Specifications and Bases, the term will be capitalized. This change is consistent with formatting requirements in the ISTS. This change is designated as administrative because it does not represent a technical change to the Technical Specifications.

A.3 CTS 1.0.A provides the definition of Alteration of the Reactor Core. ITS Section 1.1 provides a definition of CORE ALTERATION that includes an additional phrase that states "Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position." This changes the CTS by adding this phrase to the definition.

The ITS definition of CORE ALTERATION states that the suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe position. This change is acceptable because it clearly states current plant practice. The unit will not be maintained in an unsafe condition. This change is designated administrative because it represents a clarification to existing practice.

A.4 CTS Section 1.0 does not provide a definition of SHUTDOWN MARGIN (SDM). However, CTS 3.3.A.1 does specify that the core loading shall be limited to that "which can be made subcritical in the most reactive condition during the operating cycle with the strongest operable control rod in its full-out position and all other operable rods fully inserted," and CTS 4.3.A.1 specifies "that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted." ITS Section 1.1 includes a definition for SDM, which states "SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn.

DISCUSSION OF CHANGES
ITS CHAPTER 1.0, USE AND APPLICATION

With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM." This changes the CTS as follows:

- An explicit allowance has been included in the ITS Section 1.1 SDM definition to compensate for control rods which are not capable of being fully inserted.

  This change is necessary because ITS 3.1.3 allows the plant to operate with stuck control rods. This change is discussed in the Discussion of Changes for ITS 3.1.3.

- This change adds specific details defining the most reactive shutdown condition to which the SDM is analyzed; i.e., the reactor is xenon free and the moderator temperature is 68°F.

  This change is acceptable since it is consistent with current practice, as indicated in UFSAR Section 3.3.3.3, which states that shutdown capability is evaluated assuming a cold and xenon-free core. The moderator temperature used in the shutdown capability calculations assumes a moderator temperature of 68°F.

These changes are designated as administrative because they do not represent a technical change to the Technical Specifications.

A.5 CTS 1.0.D includes the definition of Immediate. It states "Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action." The ITS includes Section 1.3, "Completion Times," which describes the meaning of the term "immediately" when used as a Completion Time. It states "When "immediately" is used, the Required Action should be pursued without delay and in a controlled manner." This changes the CTS by deleting the definition of "Immediate" but adds a description to the ITS of "immediately" when used as a Completion Time.

The purpose of the CTS definition of Immediate is to ensure that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action. In the ITS, the meaning of the word "immediately" is described in ITS Section 1.3. Although the wording is not identical, the intent is the same. These changes are designated as administrative because they do not represent a technical change to the Technical Specifications.

A.6 CTS 1.0.E defines Instrument Functional Test as "the injection of a simulated signal into the primary sensor to verify proper instrument channel response, alarm, and/or initiating action." ITS Section 1.1 defines CHANNEL FUNCTIONAL TEST as "the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY" and states that the test "may be performed by means of any series of sequential, overlapping, or total channel steps." This results in a number of changes to the CTS. The addition of use of an "actual" signal is discussed in DOC L.2 while the allowance to inject the signal "as close to the sensor as practicable" in lieu of "into" the sensor is discussed in DOC L.3.

- The CTS definition states that the Instrument Functional Test shall verify "proper instrument channel response, alarm, and/or initiating action." The ITS definition states that the CHANNEL FUNCTIONAL TEST shall verify "OPERABILITY of all devices in the channel required for channel OPERABILITY."

  This change is acceptable because the statements are equivalent in that both require that the channel be verified to be OPERABLE. The CTS and the ITS use different examples of what is included in a channel, but this does not change the intent of the requirement. The ITS use of the phrase "all devices in the channel required for channel OPERABILITY" reflects the CTS understanding that the test includes only those portions of the channel needed to perform the safety function.

- The ITS definition states "The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps." The CTS definition does not include this statement.

  This change is acceptable because it states current Industry practice, and is not specifically prohibited by the CTS. This is consistent with the current implementation of the CHANNEL FUNCTIONAL TEST and does not result in a technical change to the Technical Specifications.

These changes are designated as administrative because they do not result in a technical change to the Technical Specifications.

A.7 CTS 1.0.F defines an Instrument Calibration as "the adjustment of an instrument signal output so that it corresponds, within acceptable range, accuracy, and response time to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm or trip. Response time is not part of the routine instrument calibration but will be checked once per cycle." ITS 1.0 defines a CHANNEL CALIBRATION as "the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps." This results in a number of changes to the CTS.

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- The CTS definition states "Calibration shall encompass the entire instrument including actuation, alarm or trip." The ITS definition states "The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY."

  This change is acceptable because the statements are equivalent in that both require that all needed portions of the channel be tested. The ITS definition reflects the CTS understanding that the CHANNEL CALIBRATION includes only those portions of the channel needed to perform the safety function.

- The ITS definition states that the CHANNEL CALIBRATION shall encompass the "CHANNEL FUNCTIONAL TEST." The CTS definition does not include this statement.

  This change is acceptable because the new ITS statement does not add any requirements. In both the CTS and the ITS, performance of a single test that fully meets the requirements of other tests can always be credited for satisfying the other tests.

- The ITS definition adds the statement "Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel." This allowance is not specifically stated in the CTS definition.

  The purpose of a CHANNEL CALIBRATION is to adjust the channel output so that the channel responds within the necessary range and accuracy to known values of the parameters that the channel monitors. This change is acceptable because RTDs and thermocouples are designed such that they have a fixed input/output response, which cannot be adjusted or changed once installed. Calibration of a channel containing an RTD or thermocouple is performed by applying the RTD or thermocouple fixed input/output relationship to the remainder of the channel, and making the necessary adjustments to the adjustable devices in the remainder of the channel to obtain the necessary output range and accuracy. Therefore, unlike other sensors, an RTD or thermocouple is not actually calibrated. The ITS CHANNEL CALIBRATION allowance for channels containing RTDs and thermocouples is consistent with the CTS calibration practices of these channels. It is also consistent with the allowance provided in CTS Table 4.2.1 Note (12), which states that calibration of instrument channels with RTD or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. This information is included in the ITS to avoid confusion, but does not change the current CHANNEL CALIBRATION practices for these types of channels.

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- The ITS definition states "The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps." The CTS definition does not include this statement.

  This change is acceptable because it states current Industry practice, and is not specifically prohibited by the CTS. This is consistent with the current implementation of the CHANNEL CALIBRATION and does not result in a technical change to the Technical Specifications.

- The CTS definition states that the response time is not part of the routine instrument calibration but is checked once per cycle. The ITS definition does not include this statement.

  This change is acceptable because the applicable Specifications in ITS Section 3.3 include Surveillances to cover the current response time testing requirements and the ITS includes the appropriate response time definitions.

These changes are designated as administrative because they do not result in a technical change to the Technical Specifications.

A.8 CTS Section 1.0 includes the following definitions:

- 1.0.G, Limiting Conditions for Operation (LCO);
- 1.0.I, Limiting Safety System Setting (LSSS);
- 1.0.M, Operating;
- 1.0.N, Operating Cycle;
- 1.0.P, Primary Containment Integrity;
- 1.0.Q, Protective Instrumentation Logic Definitions;
- 1.0.R, Rated Neutron Flux;
- 1.0.T, Reactor Coolant System Pressure or Reactor Vessel Pressure;
- 1.0.U, Refueling Operation and Refueling Outage;
- 1.0.V, Safety Limit;
- 1.0.W, Secondary Containment Integrity;
- 1.0.Z, Simulated Automatic Actuation;
- 1.0.AA, Transition Boiling;
- 1.0.AI, Purging;
- 1.0.AJ, Venting; and
- 1.0.AR, Allowable Value.

The ITS does not use this terminology and ITS Section 1.1 does not contain these definitions. This changes the CTS by deleting definitions that are not necessary.

These changes are acceptable because the terms are not used as defined terms in the ITS. Discussions of any technical changes related to the deletion of these terms are included in the applicable DOCs for the ITS Specifications in which the terms are dispositioned. These changes are designated as administrative because they eliminate defined terms that are no longer used.

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A.9 The CTS 1.0.J definition of Minimum Critical Power Ratio (MCPR) states that "The minimum critical power ratio is the value of critical power ratio associated with the most limiting assembly in the reactor core." In addition, the CTS 1.0.J definition states that the "Critical power ratio (CPR) is the ratio of that power in a fuel assembly which is calculated by the GEXL correlation to cause some point in the assembly to experience boiling transition to the actual assembly operating power." ITS Section 1.1 definition of MCPR states that "The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power." This changes the CTS definition of MCPR by specifying a separate MCPR is applicable to "each class of fuel" instead of a single MCPR is associated with the "most limiting assembly" and removes the explicit correlation that must be used to calculate CPR.

This change is acceptable since it will allow separate MCPRs to be monitored for each class of fuel, instead of a single, most limiting MCPR. In addition, the deletion of the specific correlation (GEXL) is acceptable since the ITS continues to require the documents that describe the appropriate analytical methods used to calculate MCPR to be listed in ITS 5.6.3, "CORE OPERATING LIMITS REPORT." These documents, which have been previously reviewed and approved by the NRC, indicate that the GEXL correlation is the approved correlation for calculating CPR (i.e., NEDE-24011-P-A, Section 1.2.5). In order to utilize a different correlation, the references listed in ITS 5.6.3 would have to be reviewed and approved by the NRC. This change is designated as administrative since there is no technical change because the MCPR is still monitored and the GEXL correlation must still be used to calculate CPR.

A.10 The CTS 1.0.L definition of Operable requires a system, subsystem, train, component, or device to be capable of performing its "specified function(s)," and requires all necessary support systems that are required for the system, subsystem, train, component, or device to perform its "function(s)" also be capable of performing their related support function(s). The ITS Section 1.1 definition of OPERABLE-OPERABILITY requires the system, subsystem, division, component, or device to be capable of performing the "specified safety function(s)," and requires all necessary support systems that are required for the system, subsystem, division, component, or device to perform its "specified safety function(s)" to also be capable of performing their related support functions. This changes the CTS by altering the requirement of the system, subsystem, etc., to be able to perform "specified function(s)" or "function(s)" to a requirement to be able to perform "specified safety function(s)."

The purpose of the CTS definition of Operable is to ensure that the safety analysis assumptions regarding equipment and variables are valid. This change is acceptable because the intent of both the CTS and ITS definitions is to address the safety function(s) assumed in the accident analysis and not encompass other non-safety functions a system, subsystem, etc., may also perform. These non-safety functions are not assumed in the safety analysis and are not needed in order to protect the public health and safety. This change is consistent with the current interpretation and use of the terms OPERABLE and

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OPERABILITY. This change is designated as administrative as it does not change the current use and application of the Technical Specifications.

A.11 The CTS 1.0.L definition of Operable requires that all necessary normal "and" emergency electrical power sources be available for the system, subsystem, train, component, or device to be OPERABLE. The ITS Section 1.1 definition of OPERABLE-OPERABILITY requires all necessary normal "or" emergency electrical power be available for the system, subsystem, etc. This changes the CTS definition of Operable by allowing a device to be considered OPERABLE with either normal or emergency power available.

The OPERABILITY requirements for normal and emergency power sources are clearly addressed in the second part to the CTS 1.0.L definition. These requirements allow only the normal or the emergency electrical power source to be OPERABLE, provided all of its redundant system(s), subsystem(s), train(s), component(s), and device(s) (redundant to the systems, subsystems, trains, components, and devices with an inoperable power source) are OPERABLE. This effectively changes the current "and" to an "or." The existing requirements (in the second part of CTS 1.0.L) are incorporated into ITS 3.8.1 ACTIONS for when a normal (offsite) or emergency (diesel generator) power source is inoperable. Therefore, the ITS definition now uses the word "or" instead of the current word "and." In ITS 3.8.1, new times are provided to perform the determination of OPERABILITY of the redundant systems. This change is discussed in the Discussion of Changes (DOCs) for ITS 3.8.1. This change is designated administrative since the ITS definition is effectively the same as the CTS definition or will be justified in the DOCs of ITS 3.8.1.

A.12 CTS Section 1.0 provides definitions for Pressure Boundary Leakage (CTS 1.0.AB), Identified Leakage (CTS 1.0.AC), Unidentified Leakage (CTS 1.0.AD), and Total Leakage (CTS 1.0.AE). ITS Section 1.1 includes these requirements in one definition called LEAKAGE and includes four categories: identified LEAKAGE; unidentified LEAKAGE; total LEAKAGE; and pressure boundary LEAKAGE. This changes the CTS by incorporating the four separate definitions into a single definition with no technical changes.

This change is acceptable because it results in no technical changes to the Technical Specifications. This change is designated an administrative change in that it rearranges existing definitions, with no change in intent.

A.13 CTS 1.0.AB states "Pressure boundary leakage shall be the leakage through a non-isolable fault in the reactor coolant system pressure boundary." ITS Section 1.1 states pressure boundary LEAKAGE is the LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) "component body, pipe wall, or vessel wall." This changes the CTS by explicitly stating the components of the RCS pressure boundary.

This change is acceptable because it results in no technical changes to the Technical Specifications. The CTS term "reactor coolant pressure boundary" is considered to be covered by the ITS phrase RCS "component body, pipe wall, or vessel wall." This change is administrative since the new definition of Pressure

Boundary LEAKAGE covers the same boundary as the CTS definition of RCS pressure boundary.

A.14 ITS Section 1.1 provides definitions of ACTIONS, AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR), LINEAR HEAT GENERATION RATE (LHGR), LOGIC SYSTEM FUNCTIONAL TEST, STAGGERED TEST BASIS, THERMAL POWER, and TURBINE BYPASS SYSTEM RESPONSE TIME. These terms are not defined in the CTS. This changes the CTS by adding the above terms.

The purpose of these ITS definitions is to define terms used in various ITS Specifications. This change is acceptable because the definitions do not impose any new requirements or alter existing requirements. Any technical changes due to the addition of these definitions will be addressed in the DOCs for the sections of the Technical Specifications in which the definitions are used. These changes are designated as administrative as they add defined terms that do not involve a technical change to the Technical Specifications.

A.15 ITS Sections 1.2, 1.3, and 1.4 contain information that is not in the CTS. This change to the CTS adds explanatory information on ITS usage that is not applicable to the CTS. The added sections are:

- <u>Section 1.2 - Logical Connectors</u>

  Section 1.2 provides specific examples of the logical connectors "<u>AND</u>" and "<u>OR</u>" and the numbering sequence associated with their use.

- <u>Section 1.3 - Completion Times</u>

  Section 1.3 provides guidance on the proper use and interpretation of Completion Times. The section also provides specific examples that aid in the use and understanding of Completion Times.

- <u>Section 1.4 - Frequency</u>

  Section 1.4 provides guidance on the proper use and interpretation of Surveillance Frequencies. The section also provides specific examples that aid in the use and understanding of Surveillance Frequency.

This change is acceptable because it aids in the understanding and use of the format and presentation style of the ITS. The addition of these sections does not add or delete technical requirements, and will be discussed specifically in those Technical Specifications where application of the added sections results in a change. This change is designated as administrative because it does not result in a technical change to the Technical Specifications.

A.16 CTS 3.1.A states that the time from initiation of any Reactor Protection System (RPS) channel trip to the de-energization of the scram pilot valve solenoids shall not exceed 50 milliseconds. ITS Section 1.1 includes a definition of REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME. The ITS definition is consistent with the CTS 3.1.A, but includes the statement "The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured." This changes the CTS by adding the sentence associated with the manner of testing. Any change to the response value of 50 milliseconds is discussed in the Discussion of Changes for ITS 3.3.1.1.

This change is acceptable because the ITS definition testing allowance is consistent with current plant practices and it is not specifically prohibited by the CTS. In addition, while Monticello is not committed to IEEE-338-1977, "Response Time Verification Tests," the definition is consistent with the guidance provided in IEEE 338-1977, Section 6.3.4. Furthermore, the results of the test are unaffected by this allowance. This change is designated as administrative as it does not result in a technical change to the response time tests.

A.17 These changes to CTS 1.0.U are provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the NRC for approval in NMC letter L-MT-04-036, from Thomas J. Palmisano (NMC) to USNRC, dated June 30, 2004. As such, these changes are administrative.

MORE RESTRICTIVE CHANGES

M.1 The CTS 1.0.A definition of Alteration of the Reactor Core applies to the act of moving any component in the region "above the core support plate, below the upper grid, and within the shroud with the vessel head removed and fuel in the vessel." The ITS Section 1.1 definition of CORE ALTERATION will only apply to the movement of fuel, sources, or reactivity control components "within the reactor vessel." This changes the CTS by expanding the region to be considered a CORE ALTERATION. The change concerning the types of "components" to be considered in the CORE ALTERATION definition is discussed in DOC L.1.

The purpose of the CORE ALTERATION definition is to assure the appropriate LCOs are being met when a CORE ALTERATION is in progress to mitigate the consequences of a reactivity excursion. This change expands the region to be considered a CORE ALTERATION from the limited region of "above the core support plate, below the upper grid, and within the shroud" to "within the reactor vessel." This change is acceptable since the applicable LCOs must now be met to limit the consequences of a reactivity excursion when any of the specified components (fuel, sources, or reactivity control components) are being moved within the reactor vessel. This will ensure the applicable LCOs are met before there is a potential to affect core reactivity. This change is designated as more restrictive because the applicable LCOs must be met when the specified components are being moved over a larger region.

M.2 CTS 1.0.A definition of Alteration of the Reactor Core exempts control rod movement using the normal drive mechanism. The ITS Section 1.1 definition of CORE ALTERATION only exempts control rod movement if there is no fuel assemblies in the associated core cell. This changes the CTS by only exempting control rod movement from the definition if there are no fuel assemblies in the associated core cell.

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The purpose of the CORE ALTERATION definition is to define components that can be moved and could result in a reactivity excursion event during refueling. Movement of a control rod whose core cell contains one or more fuel assembles could affect the reactivity of the core. Therefore, considering this type of movement a CORE ALTERATION and placing similar Technical Specification restrictions as are required for other CORE ALTERATIONS is acceptable. This change is designated as more restrictive because the applicable LCOs must be met during certain control rod movements.

M.3 CTS 1.0.K states the definition of Mode as "The reactor mode is that which is established by the mode-selector switch." CTS 1.0.B states the definition of Hot Standby as "Hot Standby means operation with the reactor critical in the startup mode at a power level just sufficient to maintain reactor pressure and temperature." CTS 1.0.O states the definition of Power Operation as "Power Operation is any operation with the mode switch in the "Start-Up" or "Run" position with the reactor critical and above 1% rated thermal power." CTS 1.0.Y states the definition of Shutdown as "The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. In this condition, a reactor scram is initiated and a rod block is inserted directly from the mode switch. The scram can be reset after a short time delay. 1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F. 2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F." ITS Section 1.1 states the definition of MODE as "A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel." In addition, a new Table (ITS Table 1.1-1) has been added that defines the actual MODES. ITS Table 1.1-1 defines the different MODES as follows:

- MODE 1 (Power Operation) is when the reactor mode switch is in the Run position;
- MODE 2 (Startup) is when the reactor mode switch is in the Refuel position and all reactor vessel head closure bolts are fully tensioned (footnote (a)) or when the reactor mode switch is in the Startup/Hot Standby position;
- MODE 3 (Hot Shutdown) is when the reactor mode switch is in the Shutdown position, all reactor vessel head closure bolts are fully tensioned (footnote (a)) and the average reactor coolant temperature is > 212°F;
- MODE 4 (Cold Shutdown) is when the reactor mode switch is in the Shutdown position, all reactor vessel head closure bolts are fully tensioned (footnote (a)) and the average reactor coolant temperature is $\leq$ 212°F; and
- MODE 5 (Refueling) is when the reactor mode switch is in the Shutdown or Refuel position and one or more reactor vessel head closure bolts are less than fully tensioned (footnote (b)).

This changes the CTS in several ways:

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- The CTS 1.0.K definition of Mode is changed by adding "average reactor coolant temperature," "reactor vessel head closure bolt tensioning specified in Table 1.1-1," and "with fuel in the reactor vessel" to the definition.

  This portion of the change is considered acceptable since the new definition is consistent with the actual ITS Table 1.1-1 requirements. Any technical changes associated with the new ITS Table 1.1-1 requirements are discussed below as part of the discussion for each of the different MODES (i.e., MODES 1 through 5). As such, this portion of the change is considered administrative but is included in this more restrictive change discussion for clarity.

- The CTS 1.0.O definition of Power Operation is being split into two distinct MODES: MODE 1 for when the reactor mode switch is in Run position; and MODE 2 for when the reactor mode switch is in the Startup/Hot Standby position. Furthermore, the reference to a power level is deleted for both MODES. Also, the CTS 1.0.B definition of Hot Standby is being combined with the MODE 2 portion of the CTS 1.0.O Power Operation definition. This changes the CTS definition such that: a. when the reactor mode switch is in Run, the unit will always be in MODE 1, even if reactor power level is < 1% rated thermal power or the reactor is subcritical; and b. when the reactor mode switch is in Startup/Hot Standby position, the unit will always be in MODE 2, even if reactor power level is < 1% rated thermal power (or just sufficient to maintain reactor pressure and temperature) or the reactor is subcritical.

  This change is acceptable since it clearly defines that MODES 1 and 2 depend on the position of the reactor mode switch, not on the power level. This ensures that the unit is always in a MODE when the reactor mode switch is placed in either the Run or Startup/Hot Standby position. Thus in the individual ITS Specifications, a CTS LCO that is applicable in the Power Operation Mode will now be required to be OPERABLE during ITS MODES 1 and 2 and a CTS LCO applicable in the CTS Hot Standby Mode (referred to as startup in the CTS LCOs) will now be required to be OPERABLE during MODE 2.

- ITS MODE 2 will now include the mode switch position of Refuel when the head closure bolts are fully tensioned (as stated in ITS Table 1.1-1 footnote (a)). Currently, this reactor mode switch and head closure bolt combination is not defined in the CTS.

  This change is considered acceptable since this is currently a plant condition that has no corresponding MODE. The new requirement will ensure proper and adequate Technical Specification requirements are applied when the reactor mode switch is in the Refuel position when all head closure bolts are fully tensioned.

- The CTS 1.0.Y definition of Shutdown is being split into two distinct MODES: MODE 3 for when the reactor mode switch is in Shutdown and (as described in part 1 of the CTS definition) the average reactor coolant temperature is

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> 212°F; and MODE 4 for when the reactor mode switch is in Shutdown and (as described in part 2 of the CTS definition) the average reactor coolant temperature is ≤ 212°F. Furthermore, for both MODE 3 and MODE 4, all reactor vessel head closure bolts must be fully tensioned. This changes the CTS definition such that all head bolts must be fully tensioned to be in either MODE 3 or 4, instead of the current requirement that no CORE ALTERATIONS are being performed.

This change is considered acceptable since it ensures that it is physically impossible to perform CORE ALTERATIONS with all head bolts fully tensioned. As a result of this change, in the individual ITS Specifications, a CTS LCO that is applicable in the Shutdown/Hot Shutdown Mode will now be required to be OPERABLE during ITS MODE 3 and a CTS LCO applicable in the CTS Shutdown/Cold Shutdown Mode will now be required to be OPERABLE during MODE 4.

- ITS MODE 5 has been added to clearly define when the unit is in the refuel mode. ITS MODE 5 is defined as the reactor mode switch in either the Shutdown or Refuel position with one or more reactor vessel head closure bolts less than full tensioned. Currently, no defined term exists in the CTS for the Refuel Mode, even though many CTS Specifications use the term Refuel Mode.

  This change is acceptable because it clearly defines when the unit is considered in the Refuel Mode. This precludes being in an undefined mode and not applying the applicable Technical Specifications when the reactor mode switch is in the Refuel position or in the Shutdown position with any reactor vessel head closure bolt not fully tensioned.

These changes are designated as more restrictive because the applicable LCOs must be met under more conditions in the ITS as compared to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 1 - Removing Details of System Design and System Description, Including Design Limits)* The CTS 1.0.Y definition of Shutdown states that with the reactor mode switch in Shutdown, "a reactor scram is initiated...directly from the mode switch. The scram can be reset after a short time delay." ITS Table 1.1-1 does not include this additional design information. This changes the CTS by moving the functional description and logic associated with the reactor mode switch scram to the ITS 3.3.1.1 Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS (ITS 3.3.1.1) still retains the requirement that the reactor mode switch scram be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 CTS 1.0.A definition of Alteration of the Reactor Core applies to the act of moving "any component." However, the definition also states that the normal operating functions such as control rod movement using the normal drive mechanism, tip scans, SRM and IRM detector movements, etc., are not to be considered core alterations. The ITS Section 1.1 definition of CORE ALTERATION will only apply to the movement of "fuel, sources, or reactivity control components." In addition, the following exceptions are not considered to be CORE ALTERATIONS in the ITS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. control rod movement, provided there are no fuel assemblies in the associated core cell. This changes the CTS by eliminating from the definition of Alteration of the Core the movement of components that do not affect the reactivity of the core i.e., that are not fuel, sources, or reactivity control components, and also explicitly excludes local power range monitors and special moveable detectors from being a CORE ALTERATION. The change in the control rod movement portion of the definition is discussed in DOC M.2.

The purpose of the CORE ALTERATION definition is to define components that can be moved and could result in a reactivity excursion event during refueling. This change eliminates the movement of components that do not affect the reactivity of the core from the definition. This change is acceptable because the ITS definition of CORE ALTERATION and the associated Specifications which require equipment to be OPERABLE or parameters be met during a CORE ALTERATION will help ensure the proper controls during the movement of components such as fuel, sources, and reactivity control components. The movement of these components may affect the core reactivity, therefore these controls are necessary. Movement of local power range monitors, special movable detectors, and control rods with no fuel in the associated core cells are explicitly excluded from the definition since the movement of these components does not affect the reactivity of the core. This change is designated as less restrictive because the ITS definition of CORE ALTERATION applies in fewer circumstances than does the CTS definition.

L.2 The CTS 1.0.E definition of Instrument Functional Test requires the use of a "simulated" signal when performing the test. The ITS Section 1.1 CHANNEL FUNCTIONAL TEST definition allows the use of a "simulated or actual" signal when performing the test. This changes the CTS by allowing the use of unplanned actuations to perform the Surveillance if sufficient information is collected to satisfy the surveillance test requirements.

This change is acceptable because the channel itself cannot discriminate between an "actual" or "simulated" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change is designated as less restrictive because it allows an actual signal to be credited for a Surveillance where only a simulated signal was previously allowed.

L.3 CTS 1.0.E defines Instrument Functional Test as the injection of a simulated signal "into the primary sensor." ITS Section 1.1 defines CHANNEL FUNCTIONAL TEST as the injection of a simulated or actual signal "into the channel as close to the sensor as practicable." This changes the CTS by allowing a signal to be injected "in the channel as close to the sensor as practicable" instead of "into the primary sensor."

The purpose of a CHANNEL FUNCTIONAL TEST is to ensure a channel is OPERABLE. This change allows a CHANNEL FUNCTIONAL TEST to be performed by injecting a signal "as close to the sensor as practicable" instead of "into the primary sensor." Injecting a signal into the primary sensor would, in some cases, involve significantly increased probabilities of initiating undesired circuits during the test since several logic channels are often associated with a particular sensor. Performing the test by injection of a signal into the primary sensor could also require jumpering of the other logic channels to prevent their initiation during the test or increasing the scope of the tests to include multiple tests of the other logic channels. Either method significantly increases the difficulty of performing the surveillance. Allowing initiation of the signal close to the sensor in lieu of into the sensor provides a complete test of the logic channel while significantly reducing the probability of undesired initiation. In addition, the sensor is still being checked during a CHANNEL CALIBRATION. This change is designated as less restrictive because the ITS definition of CHANNEL FUNCTIONAL TEST will allow the test to be performed injecting a signal "into the channel as close to the sensor as practicable" instead of "into the primary sensor."

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

1.0 USE AND APPLICATION

CTS

1.1 Definitions

DOC A.2

---NOTE---

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

	Term	Definition	
DOC A.14	ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.	
DOC A.14	AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	The APLHGR shall be applicable to a specific planar height and is equal to the sum of the [LHGRs] [heat generation rate per unit length of fuel rod] for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle [at the height].	1 1
1.0.F	CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.	
1.0.X	CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.	
1.0.E	CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.	

CTS 1.1 Definitions

1.0.A	CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement), and ; (2) b. Control rod movement, provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
1.0.AQ	CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.[5]. ~~Plant operation within these limits is addressed in individual Specifications.~~ [3] (TSTF-369)
1.0.AK	DOSE EQUIVALENT I-[1]31	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in [Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, ~~or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity"~~]. (1) (3) (1)
	~~EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME~~	~~The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the~~ (6)

CTS 1.1 Definitions

ECCS RESPONSE TIME (continued)	
	entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC. (6)
END OF CYCLE RECIRCULATION PUMP TRIP (EOC RPT) SYSTEM RESPONSE TIME	The EOC RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by [the associated turbine stop valve limit switch or from when the turbine control valve hydraulic oil control oil pressure drops below the pressure switch setpoint] to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured, [except for the breaker arc suppression time, which is not measured but is validated to conform to the manufacturer's design value]. (4)
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC. (6)

DOC A.12 LEAKAGE — LEAKAGE shall be:

a. Identified LEAKAGE

1.0.AC 1. LEAKAGE into the drywell, such as that from pump seals or valve packing that is captured and conducted to a sump or collecting tank, or ; (2)

1.0.AC 2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE, ; (2)

CTS 1.1 Definitions

LEAKAGE (continued)

1.0.AD b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE[.] ; (2)

1.0.AE c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE[,] and ; (2)

1.0.AB d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

DOC A.14 [LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.] (1)
DOC A.14 LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
~~[MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)~~	~~The MFLPD shall be the largest value of the fraction of limiting power density in the core. The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.]~~ (4)
1.0.J MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core [for each class of fuel]. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power. (1)
1.0.K MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

CTS

1.1 Definitions

1.0.L	OPERABLE – OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).	
	~~PHYSIS TESTS~~	~~PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation.~~ ~~These tests are:~~ ~~a. Described in Chapter [14, Initial Test Program] of the FSAR,~~ ~~b. Authorized under the provisions of 10 CFR 50.59, or~~ ~~c. Otherwise approved by the Nuclear Regulatory Commission.~~	4
	~~PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~	~~The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.~~$\boxed{6}$ 4	7 TSTF -369
1.0.S	RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~[2436]~~ 1775 MWt.	1
3.1.A	REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from ~~when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until~~ initiation of any RPS channel trip to the de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. ~~In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.~~	5 5

CTS 1.1 Definitions

3.3.A.1	SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free ; 2 b. The moderator temperature is 68°F and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
DOC A.14	STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
DOC A.14	THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
DOC A.14	[TURBINE BYPASS SYSTEM RESPONSE TIME shall be that time interval from when the main turbine trip solenoid is activated	The TURBINE BYPASS SYSTEM RESPONSE TIME ~~consists of two components:~~ a. ~~The time from initial movement of the main turbine stop valve or control valve~~ until 80% of the turbine bypass capacity is established ~~and~~ . b. ~~The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve.~~ The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.] 8

Table 1.1-1 (page 1 of 1)
MODES

CTS

CTS	MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1.0.O	1	Power Operation	Run	NA
1.0.O	2	Startup	Refuel[(a)] or Startup/Hot Standby	NA
1.0.Y.1	3	Hot Shutdown[(a)]	Shutdown	> ~~[200]~~ 212
1.0.Y.2	4	Cold Shutdown[(a)]	Shutdown	≤ ~~[200]~~ 212
DOC M.3	5	Refueling[(b)]	Shutdown or Refuel	NA

1.0.O, 1.O.Y (a) All reactor vessel head closure bolts fully tensioned.

DOC M.3 (b) One or more reactor vessel head closure bolts less than fully tensioned.

1.0 USE AND APPLICATION

CTS

1.2 Logical Connectors

DOC A.15 PURPOSE	The purpose of this section is to explain the meaning of logical connectors. Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.
BACKGROUND	Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors. When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.
EXAMPLES	The following examples illustrate the use of logical connectors.

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CTS 1.2 Logical Connectors

DOC A.15 EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

CTS 1.2 Logical Connectors

DOC A.15 EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

CTS

DOC A.15

PURPOSE

The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND

Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

Once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

However, when a subsequent division, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability and ; 2

b. Must remain inoperable or not within limits after the first inoperability is resolved.

CTS 1.3 Completion Times

DOC A.15 DESCRIPTION (continued)

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours ; or 2

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each division, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." ~~Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.~~ 9

EXAMPLES The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

CTS 1.3 Completion Times

DOC A.15 EXAMPLES (continued)

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 12 hours AND in MODE 4 within 36 hours. A total of 12 hours is allowed for reaching MODE 3 and a total of 36 hours (not 48 hours) is allowed for reaching MODE 4 from the time that Condition B was entered. If MODE 3 is reached within 6 hours, the time allowed for reaching MODE 4 is the next 30 hours because the total time allowed for reaching MODE 4 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 4 is the next 36 hours.

CTS 1.3 Completion Times

DOC A.15 EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

CTS 1.3 Completion Times

DOC A.15 EXAMPLES (continued)

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X subsystem inoperable.	A.1 Restore Function X subsystem to OPERABLE status.	7 days ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~ 9
B. One Function Y subsystem inoperable.	B.1 Restore Function Y subsystem to OPERABLE status.	72 hours ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~ 9
C. One Function X subsystem inoperable. AND One Function Y subsystem inoperable.	C.1 Restore Function X subsystem to OPERABLE status. OR C.2 Restore Function Y subsystem to OPERABLE status.	72 hours 72 hours

CTS 1.3 Completion Times

DOC A.15 EXAMPLES (continued)

When one Function X subsystem and one Function Y subsystem are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each subsystem starting from the time each subsystem was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second subsystem was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected subsystem was declared inoperable (i.e., initial entry into Condition A).

~~The Completion Times of Conditions A and B are modified by a logical connector with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.~~

INSERT 1

9

9

INSERT 1

It is possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. However, doing so would be inconsistent with the basis of the Completion Times. Therefore, there shall be administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls shall ensure that the Completion Times for those Conditions are not inappropriately extended.

CTS 1.3 Completion Times

DOC A.15 EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered.

CTS 1.3 Completion Times

DOC A.15 EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

--------------------NOTE--------------------
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

CTS 1.3 Completion Times

DOC A.15 EXAMPLES (continued)

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

CTS 1.3 Completion Times

DOC A.15 EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 4.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

CTS

1.3 Completion Times

1.0.D	IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

CTS

DOC A.15

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0[2], Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements. 3

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied: 3

CTS

1.4 Frequency

DOC A.15

DESCRIPTION (continued)

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; $\boxed{or}$ 3

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Examples 1.4-3 and 1.4-4), then SR 3.0.3 becomes applicable.

CTS 1.4 Frequency

DOC A.15 EXAMPLES (continued)

then SR 3.0.4 becomes applicable.

, as modified by SR 3.0.3,

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. ~~Failure to do so would result in a violation of SR 3.0.4.~~

or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

10

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

CTS 1.4 Frequency

DOC A.15 EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-------------------------NOTE------------------------- Not required to be performed until 12 hours after ≥ 25% RTP. --	
Perform channel adjustment.	7 days

The interval continues, whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance ~~were~~ was not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP. 3

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance ~~were~~ not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

CTS

1.4 Frequency

DOC A.15

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----NOTE----- Only required to be met in MODE 1. -----	
Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance ~~were~~ was not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency ~~were~~ was not met), SR 3.0.4 would require satisfying the SR. 3

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----NOTE----- Only required to be performed in MODE 1. -----	
Perform complete cycle of the valve.	7 days

The interval continues, whether or not the unit operation is in MODE 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

CTS 1.4 Frequency

DOC A.15 EXAMPLES (continued)

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. (3) was
Therefore, if the Surveillance ~~were~~ not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance ~~were~~ not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-------------------------NOTE------------------------- Not required to be met in MODE 3. --	
Verify parameter is within limits.	24 hours

Example 1.4-[6] specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance ~~were~~ not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, (1) was (3)

CTS 1.4 Frequency

DOC A.15 EXAMPLES (continued)

even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency ~~were~~ was not met), SR 3.0.4 would require satisfying the SR. 3

JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

1. The brackets are removed and the proper plant specific information/value is provided.

2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

3. Typographical/grammatical error corrected.

4. The definitions of EOC-RPT SYSTEM RESPONSE TIME, MAXIMUM FRACTION OF LIMITING POWER DENSITY, and PHYSICS TESTS have been deleted since they are not used in the Monticello ITS.

5. The current licensing basis definition for the RPS RESPONSE TIME has been maintained.

6. ECCS RESPONSE TIME and ISOLATION SYSTEM RESPONSE TIME definitions have not been adopted, consistent with Monticello current licensing basis. Monticello response time requirements reflect the industry standards and regulations to which the plant has been committed to and licensed to since the operating license was granted. Monticello is committed to the testing requirements contained in IEEE-279-1968 and IEEE-338-1971. These industry standards provide guidance and requirements for conducting periodic testing of protection systems. IEEE-279-1968 does not address response time testing. Response time testing requirements do not appear in IEEE-338 until the 1975 revision.

7. Monticello does not propose to use a PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) and will not relocate the Pressure and Temperature limits from the Technical Specifications. The current limits will be retained in the ITS. Therefore, the definition of PTLR has not been incorporated into the ITS.

8. The brackets are removed and the proper plant specific information is provided. This is consistent with current transient analysis assumptions, plant design, and the manner in which the TURBINE BYPASS SYSTEM RESPONSE TIME is currently measured.

9. These changes are made consistent with TSTF-439, Rev. 2, which has been incorporated by the USNRC into Revision 3.1 of NUREG-1433.

10. These changes are made consistent with TSTF-485, Rev. 0, which has been approved by the USNRC for incorporation into Revision 3.1 of NUREG-1433 as documented in a letter from T. H. Boyce (NRC) to the Technical Specifications Task Force, dated 12/6/05.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.1

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1433.

CTS 1.0.A definition of Alteration of the Reactor Core applies to the act of moving "any component." However, the definition also states that the normal operating functions such as control rod movement using the normal drive mechanism, tip scans, SRM and IRM detector movements, etc., are not to be considered core alterations. The ITS Section 1.1 definition of CORE ALTERATION will only apply to the movement of "fuel, sources, or reactivity control components." In addition, the following exceptions are not considered to be CORE ALTERATIONS in the ITS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. control rod movement, provided there are no fuel assemblies in the associated core cell. This changes the CTS by eliminating from the definition of Alteration of the Core the movement of components that do not affect the reactivity of the core i.e., that are not fuel, sources, or reactivity control components, and also explicitly excludes local power range monitors, special moveable detectors, and control rods with no fuel in the associated core cells from being a CORE ALTERATION. The change in the control rod movement portion of the definition is discussed in DOC M.2.

The purpose of the CORE ALTERATION definition is to define components that can be moved and could result in a reactivity excursion event during refueling. This change eliminates the movement of components that do not affect the reactivity of the core from the definition. This change is acceptable because the ITS definition of CORE ALTERATION and the associated Specifications which require equipment to be OPERABLE or parameters be met during a CORE ALTERATION will help ensure the proper controls during the movement of components such as fuel, sources, and reactivity control components. The movement of these components may affect the core reactivity, therefore these controls are necessary. Movement of local power range monitors and special movable detectors are explicitly excluded from the definition since the movement of these components does not affect the reactivity of the core. This change is designated as less restrictive because the ITS definition of CORE ALTERATION applies in fewer circumstances than does the CTS definition.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

The proposed change revises the definition of CORE ALTERATION to be only the movement of fuel, sources, or reactivity control components rather than the movement of any component, and also explicitly excludes local power range monitors, special moveable detectors, and control rods with no fuel in the associated core cells. This change will not affect the probability of an accident. The only component assumed to be an initiator of an event previously evaluated is an irradiated fuel assembly when it is dropped. None of the other components are initiators of any analyzed event. As fuel is retained in the list of components which, when moved, constitutes a CORE ALTERATION, the probability of a fuel handling accident is not affected. The consequences of an accident are not affected by this change as movement of the components being excluded from the definition of CORE ALTERATION do not act to mitigate the consequences of any accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change revises the definition of CORE ALTERATION to be the movement of fuel, sources, or reactivity control components rather than the movement of any component, and also explicitly excludes local power range monitors, special moveable detectors, and control rods with no fuel in the associated core cells. This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with current safety analysis assumptions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change revises the definition of CORE ALTERATION to be the movement of fuel, sources, or reactivity control components rather than the movement of any component, and also explicitly excludes local power range monitors, special moveable detectors, and control rods with no fuel in the associated core cells. The margin of safety is not affected by this change because the safety analysis assumptions are not affected. The safety analyses do not address the movement of components within the reactor vessel other than fuel and reactivity control components. Fuel continues to be included in the CORE ALTERATION definition. Also, the SHUTDOWN MARGIN is unaffected by the movement of components other than fuel, sources, and reactivity control components because the movement of other components will not significantly change core reactivity. No change is being proposed in the application of the definition to the movement of components that are factors in the design basis analyses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.2

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1433.

The CTS 1.0.E definition of Instrument Functional Test requires the use of a "simulated" signal when performing the test. The ITS Section 1.1 CHANNEL FUNCTIONAL TEST definition allows the use of an "simulated or actual" signal when performing the test. This changes the CTS by allowing the use of unplanned actuations to perform the Surveillance if sufficient information is collected to satisfy the surveillance test requirements.

This change is acceptable because the channel itself cannot discriminate between an "actual" or "simulated" signal and, therefore, the results of the testing are unaffected by the type of signal used to initiate the test. This change is designated as less restrictive because it allows an actual signal to be credited for a Surveillance where only a simulated signal was previously allowed.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change adds an allowance that an actual as well as a simulated signal can be credited during the CHANNEL FUNCTIONAL TEST. This change allows taking credit for unplanned actuations if sufficient information is collected to satisfy the surveillance test requirements. This change is acceptable because the channel itself cannot discriminate between an "actual" or "simulated" signal, and the proposed requirement does not change the technical content or validity of the test. This change will not affect the probability of an accident. The source of the signal sent to components during a Surveillance is not assumed to be an initiator of any analyzed event. The consequence of an accident is not affected by this change. The results of the testing, and, therefore, the likelihood of discovering an inoperable component, are unaffected. As a result, the assurance that equipment will be available to mitigate the consequences of an accident is unaffected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No.

The proposed change adds an allowance that an actual as well as a simulated signal can be credited during the CHANNEL FUNCTIONAL TEST. This change will not physically alter the plant (no new or different type of equipment will be installed). The change also does not require any new or revised operator actions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

The proposed change adds an allowance that an actual as well as a simulated signal can be credited during the CHANNEL FUNCTIONAL TEST. The margin of safety is not affected by this change. This change allows taking credit for unplanned actuations if sufficient information is collected to satisfy the surveillance test requirements. This change is acceptable because the channel itself cannot discriminate between an "actual" or "simulated" signal. As a result, the proposed requirement does not change the technical content or validity of the test. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.3

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1433.

CTS Section 1.0 defines Instrument Functional Test as the injection of a simulated signal "into the primary sensor." ITS Section 1.1 defines CHANNEL FUNCTIONAL TEST as the injection of a simulated or actual signal "into the channel as close to the sensor as practicable." This changes the CTS by allowing a signal to be injected "in the channel as close to the sensor as practicable" instead of "into the primary sensor."

The purpose of a CHANNEL FUNCTIONAL TEST is to ensure a channel is OPERABLE. This change allows a CHANNEL FUNCTIONAL TEST to be performed by injecting a signal "as close to the sensor as practicable" instead of "into the primary sensor." Injecting a signal into the primary sensor would, in some cases, involve significantly increased probabilities of initiating undesired circuits during the test since several logic channels are often associated with a particular sensor. Performing the test by injection of a signal into the primary sensor could also require jumpering of the other logic channels to prevent their initiation during the test or increasing the scope of the tests to include multiple tests of the other logic channels. Either method significantly increases the difficulty of performing the surveillance. Allowing initiation of the signal close to the sensor in lieu of into the sensor provides a complete test of the logic channel while significantly reducing the probability of undesired initiation. In addition, the sensor is still being checked during a CHANNEL CALIBRATION. This change is designated as less restrictive because the ITS definition of CHANNEL FUNCTIONAL TEST will allow the test to be performed injecting a signal "into the channel as close to the sensor as practicable" instead of "into the primary sensor."

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   Testing of instrument channels such that the test signal does not include the "sensor" will significantly reduce the complications associated with performance of a surveillance on a sensor that provides input to multiple logic channels. The sensor will still be checked during a CHANNEL CALIBRATION. This reduction of complication will not affect the failure probability of the equipment but may reduce the probability of personnel error during the surveillance. Such reductions will not involve a significant increase in the probability or consequences of an accident previously evaluated.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 1.0, USE AND APPLICATION

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   This change does not involve a change to the limits or limiting condition of operation; only the method for performing a surveillance is changed. Since the proposed method affects only a single logic channel rather than potentially affecting multiple logic channels simultaneously, and the sensor is adequately tested during a CHANNEL CALIBRATION, the change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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Summary of Changes
ITS Chapter 2.0

Change Description	Affected Pages
The changes described in the NMC response to Question 200601201446 have been made. Minor editorial changes are made.	Pages 17 and 19 of 24

ATTACHMENT 1

VOLUME 4

MONTICELLO
IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS CHAPTER 2.0
SAFETY LIMITS

Revision 1

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS Chapter 2.0, Safety Limits

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

Attachment 1, Volume 4, Rev. 1, Page 5 of 24

2.0 SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS
2.1 — 2.1 SAFETY LIMITS	~~Limiting Safety System Settings are Incorporated into Section 3 of the Technical Specifications.~~

2.1.1 A. Reactor Core Safety Limits

2.1.1.1 1. With the reactor steam dome pressure <785 psig or core flow $<10\%$ rated core flow:

Thermal power shall be $\leq 25\%$ Rated Thermal Power

2.1.1.2 2. With the reactor steam dome pressure ≥ 785 psig and core flow $\geq 10\%$ rated core flow:

MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

2.1.1.3 3. Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 B. Reactor Coolant System Pressure Safety Limit

Reactor steam dome pressure shall be ≤ 1332 psig.

Attachment 1, Volume 4, Rev. 1, Page 5 of 24

2.0 SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS
2.2 SAFETY LIMIT VIOLATIONS	
2.2 With any Safety Limit violation, the following actions shall be completed within 2 hours:	
2.2.1 A. Restore compliance with all Safety Limits; and	
2.2.2 B. Insert all insertable control rods.	

Attachment 1, Volume 4, Rev. 1, Page 6 of 24

DISCUSSION OF CHANGES
ITS CHAPTER 2.0, SAFETY LIMITS

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

2.0 SAFETY LIMITS (SLs)

2.1 **2.1 SLs**

2.1.A 2.1.1 Reactor Core SLs

2.1.A.1 2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.A.2 2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.10 ~~[1.07]~~ for two recirculation loop operation or ≥ 1.12 ~~[1.08]~~ for single recirculation loop operation. 1

2.1.A.3 2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.B 2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1332 ~~1325~~ psig. 2

2.2 SL VIOLATIONS

2.2 With any SL violation, the following actions shall be completed within 2 hours:

2.2.A 2.2.1 Restore compliance with all SLs; and

2.2.B 2.2.2 Insert all insertable control rods.

JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 2.0, SAFETY LIMITS

1. The brackets are removed and the proper plant specific information/value is provided.

2. Changes have been made to reflect the current licensing basis value.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

INSERT 1

BASES

BACKGROUND

~~GDC 10 (Ref. 1) requires, and~~ SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). 1

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2 ~~for [both General Electric Company (GE) and Advanced Nuclear Fuel Corporation (ANF) fuel]~~. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. 2

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant. INSERT 1A 4

1 **INSERT 1**

USAR Section 1.2.2 (Ref. 1) requires the reactor core and associated systems to be designed to accommodate plant operational transients or maneuvers that might be expected without compromising safety and without fuel damage. Therefore,

4 **INSERT 1A**

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this SL provides margin such that the SL will not be reached or exceeded.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that a~~n~~ MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling. 5

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR Safety limit. 6

2.1.1.1~~a~~ Fuel Cladding Integrity ~~[General Electric Company (GE) Fuel]~~ 2

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 psig and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > ~~4.5~~ 6 psi. Analyses (Ref. 2) show that with a bundle flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a ~~4.5~~ psi driving head will be > 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from ~~14.7 psia~~ 0 psig to ~~800 psia~~ 785 psig indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50 % RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig or < 10% core flow is conservative. 1 1 1 5

~~2.1.1.1b Fuel Cladding Integrity [Advanced Nuclear Fuel Corporation (ANF) Fuel]~~

~~The use of the XN-3 correlation is valid for critical power calculations at pressures > 580 psig and bundle mass fluxes > 0.25 x 10^6 lb/hr-ft^2 (Ref. 3). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:~~ 2

~~Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition. For the ANF 9x9 fuel design, the minimum bundle flow is > 30 x 10^3 lb/hr. For the ANF 8x8 fuel design, the minimum bundle flow is > 28 x 10^3 lb/hr. For all designs, the coolant minimum bundle flow and maximum flow area are~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~such that the mass flux is always > 0.25 x 10^6 lb/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 25% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of > 3.0, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% RTP for reactor pressures < 785 psig is conservative.~~ 2

2.1.1.2a MCPR ~~[GE Fuel]~~ 2

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference ~~2 also~~ 3 includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis. 1

~~2.1.1.2b MCPR [ANF Fuel]~~

~~The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical~~ 2

BASES

APPLICABLE SAFETY ANALYSES (continued)

procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the XN-3 critical power correlation. Reference 3 describes the methodology used in determining the MCPR SL.

The XN-3 critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the XN-3 correlation, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 correlation provide a reasonable degree of assurance that there would be no transition boiling in the core during sustained operation at the MCPR SL. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition. (2)

2.1.1.3 Reactor Vessel Water Level

Irradiated

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action. (6)

BASES

SAFETY LIMITS	The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations. 5
APPLICABILITY	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.
SAFETY LIMIT VIOLATIONS	Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.
REFERENCES	1. ~~10 CFR 50, Appendix A, GDC 10.~~ USAR, Section 1.2.2. 1 2. NEDE-24011-P-A ~~(latest approved revision)~~, "General Electric Standard Application for Reactor Fuel" (revision specified in Specification 5.6.3). 6 3. ~~XN-NF524(A), Revision 1, November 1983.~~ NEDE-31152P, "General Electric Fuel Bundle Designs," Revision 8, April 2001. 1 4. 10 CFR 100.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. ~~According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).~~ (1)

INSERT 2

for the pressure vessel

anticipated operational occurrences (AOOs)

During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3). (1) (1)

reactor coolant pressure boundary (

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere. (6)

)

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure - High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

1965

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME, Boiler and Pressure Vessel Code, ~~[1971]~~ Edition~~]~~, including Addenda through the ~~[winter of 1972]~~ (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. ~~The SL~~ of ~~[1325]~~ psig, as measured in the reactor steam dome, is (3) (1)

summer of 1966

1335

A pressure

1 **INSERT 2**

According to USAR Section 4.2.1 (Ref. 1), the reactor vessel design pressure of 1250 psig was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation, with additional allowances to accommodate transients above the operating pressure without causing operation of the safety/relief valves. In addition, the reactor vessel was also designed for the transients that could occur during the design life.

BASES

APPLICABLE SAFETY ANALYSES (continued) summer of 1978

equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to the USAS Nuclear Power Piping Code, Section B31.1,
1977 [~~1969~~ Edition], including Addenda through [~~July 1, 1970~~] (Ref. 6), for the 3
~~reactor recirculation~~ piping, which permits a maximum pressure transient
120 of [~~110~~]% of design pressures of [1250 psig ~~for suction~~ piping] and INSERT 3 1
INSERT 4 1500 psig ~~for discharge~~ piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The
120 maximum transient pressure allowable in the RCS piping, valves, and INSERT 5
fittings is [~~110~~]% of design pressures of [1250 psig ~~for suction~~ piping and 1
1500 psig ~~for discharge~~ piping]. The most limiting of these allowances is
120 the [~~110~~]% of the [~~suction~~] piping design pressure[s]; therefore, the SL on 1332
communicating with the vessel steam space maximum allowable RCS pressure is established at [~~1325~~] psig as 1
measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. [10 CFR 50, Appendix A, ~~GDC 14~~, GDC 15, and GDC 28] 1
USAR, Section 4.2.1

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.

4. 10 CFR 100.

1965
5. ASME, Boiler and Pressure Vessel Code, Section III, [~~1971~~] Edition[], Addenda [~~winter of 1972~~]. 3
summer of 1966 1977

6. ASME, USAS, Nuclear Power Piping Code, Section B31.1, [~~1969~~] Edition[], Addenda [~~July 1, 1970~~]. 3
summer 1978

1

INSERT 3

1110 psig for piping communicating with the vessel steam space

1

INSERT 4

1136 psig for piping communicating with the bottom of the vessel

1

INSERT 5

1110 psig for piping communicating with the vessel steam space and 1136 psig for piping communicating with the bottom of the vessel

JUSTIFICATION FOR DEVIATIONS
ITS CHAPTER 2.0 BASES, SAFETY LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases, which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The Bases has been modified to reflect the fuel used at Monticello. The Monticello reactor core does not contain Advanced Nuclear Fuel Corporation (ANF) Fuel.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. A description of the reactor vessel water level SL has been added, consistent wih the Background description of the other SLs.

5. Typographical/grammatical error corrected.

6. Editorial change made for clarity.

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Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS CHAPTER 2.0, SAFETY LIMITS

There are no specific NSHC discussions for this Specification.

Summary of Changes
ITS Section 3.0

Change Description	Affected Pages
The changes described in the NMC response to Question 200510031651 have been made. Typographical correction to Bases JFD 7 has been made.	Page 55 of 69
The changes described in the NMC response to Question 200512151125 have been made. Changes are made to be consistent with TSTF-372, Rev. 4 (Addition of LCO 3.0.8, Inoperability of Snubbers). In addition, the last paragraph of ITS Bases INSERT 1 has been modified by deleting the words "train or", consistent with the deletion of these words in the other paragraphs of the ITS Bases INSERT.	Pages 11, 12, 20, 21, 29, 31, 32, 35, 37, 44, 46, 47, 48, and 55 of 69
The changes described in the NMC response to Question 200601201446 have been made. Minor markup correction to the Bases Markup has been made.	Page 42 of 69

ATTACHMENT 1

VOLUME 5

MONTICELLO
IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.0
LCO and SR APPLICABILITY

Revision 1

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS Section 3.0, LCO and SR Applicability

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

(LCO) APPLICABILITY

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	3.0	4.0 SURVEILLANCE REQUIREMENTS

(SR) APPLICABILITY

3.0 4.0 SURVEILLANCE REQUIREMENTS

SRs

INSERT 7

SR 3.0.1 A. The surveillance requirements of this section shall be met. Each surveillance requirement shall be performed at the specified times except as allowed in B and C below. A.5

SR 3.0.2 B. Specific time intervals between tests may be extended up to 25% of the surveillance interval to accommodate normal test schedules with the exception that, the intervals between tests scheduled for refueling shutdowns shall not exceed two years. INSERT 8 M.2 A.6 LA.1 A.7 L.3

SR 3.0.1 C. Whenever the plant condition is such that a system or component is not required to be operable the surveillance testing associated with that system or component may be discontinued.

SR 3.0.4 Discontinued surveillance tests shall be resumed less than one test interval before establishing plant conditions requiring operability of the associated system or component. A.5 INSERT 9 L.4

SR 3.0.1 D. If it is discovered that a surveillance was not performed within the extended time interval allowed by 4.0.B, then the affected equipment shall be declared inoperable. A.5

A.1 INSERT 10

SR 3.0.3 E. Compliance with 4.0.D may be delayed, from the time of discovery, up to 24 hours or up to the limit of the time interval, whichever is greater. This delay period is permitted to allow performance of the surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

specified Frequency

the requirement to declare the LCO not met

A.1

INSERT 11

A.2 INSERT 1

M.1 INSERT 2

L.1 INSERT 3

L.2 INSERT 4

A.3 INSERT 5

A.4 INSERT 6

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Attachment 1, Volume 5, Rev. 1, Page 5 of 69

A.2 **INSERT 1**

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

M.1 **INSERT 2**

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 2 within 7 hours;

b. MODE 3 within 13 hours; and

c. MODE 4 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, and 3.

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L.1 **INSERT 3**

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications; or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

L.2 **INSERT 4**

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

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A.3 **INSERT 5**

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.10, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

A.4 **INSERT 6**

LCO 3.0.7 Special Operations LCOs in Section 3.10 allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Special Operations LCOs is optional. When a Special Operations LCO is desired to be met but is not met, the ACTIONS of the Special Operations LCO shall be met. When a Special Operations LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

A.5 **INSERT 7**

during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

INSERT 8

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. A.6

For Frequencies specified as "once," the above interval extension does not apply. M.2

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. L.3

Exceptions to this Specification are stated in the individual Specifications. A.6

L.4

INSERT 9

SR 3.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

A.1

INSERT 10

If it is discovered that a Surveillance was not performed within its specified Frequency, then

A.1

INSERT 11

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
H. Snubbers 1. Except as permitted below, all safety related snubbers shall be operable whenever the supported system is required to be Operable. LCO 3.0.8 2. With one or more snubbers made or found to be inoperable for any reason when Operability is required, within 72 hours: a. Replace or restore the inoperable snubbers to Operable status and perform an engineering evaluation or inspection of the supported components, or b. Determine through engineering evaluation that the as-found condition of the snubber had no adverse effect on the supported components and that they would retain their structural integrity in the event of design basis seismic event, or c. Declare the supported system inoperable and take the action required by the Technical Specifications for inoperability of that system.	H. Snubbers The following surveillance requirements apply to all safety related snubbers. 1. Visual inspections: Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible or accessible) may be inspected independently according to the schedule determined by Table 4.6-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.6-1. The initial inspection interval for new types of snubbers shall be established at 18 months +25%. INSERT 12 M.3 See CTS 3/4.6.H

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M.3

INSERT 12

LCO 3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. The snubbers not able to perform their associated support function(s) are associated with only one subsystem of a multiple subsystem supported system or are associated with a single subsystem supported system and are able to perform their associated support function within 72 hours; or

b. The snubbers not able to perform their associated support function(s) are associated with more than one subsystem of a multiple subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

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DISCUSSION OF CHANGES
ITS SECTION 3.0, LCO AND SR APPLICABILITY

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 ITS LCO 3.0.1 and LCO 3.0.2 are added to the CTS to provide guidance regarding LCOs and ACTIONS. ITS LCO 3.0.1 states "LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7." ITS LCO 3.0.2 states "Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated." The changes to the CTS are:

- CTS 3/4.0 does not include any general LCO/ACTION guidance requirements. However, in general the CTS LCOs require either the equipment to be OPERABLE or parameters to be met during the specified conditions. This is consistent with ITS LCO 3.0.1. In addition, if the LCO is not met, the applicable CTS Specification provides the appropriate actions to take. ITS LCO 3.0.2 states, in part, "Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met." This statement is consistent with the current application of CTS actions. The second sentence of ITS LCO 3.0.2 states, in part, "If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required." This statement is also consistent with the current application of the CTS actions. The second sentence of ITS LCO 3.0.2 includes the phrase, "unless otherwise stated" at the end of the sentence. There are some ITS ACTIONS, which must be completed, even if the LCO is met or is no longer applicable. While this is a new requirement, the technical aspects of these changes are discussed in the appropriate ITS Specifications.

  This change is acceptable because the intent of the CTS requirements is preserved and results in no technical changes to the Technical Specifications.

- LCO 3.0.2 includes exceptions for LCO 3.0.5 and LCO 3.0.6. LCO 3.0.5 is a new allowance, for a system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY, that takes exception to the ITS LCO 3.0.2 requirement. LCO 3.0.6 is a new allowance that takes exception to the ITS LCO 3.0.2 requirement to take the Required Actions of a supported system LCO when the inoperability is

DISCUSSION OF CHANGES
ITS SECTION 3.0, LCO AND SR APPLICABILITY

only associated with a support system LCO. These exceptions are included in LCO 3.0.2 to avoid conflicts between the applicability requirements.

This change is acceptable because it includes a reference to new items in the ITS. Changes resulting from the incorporation of LCO 3.0.5 are discussed in DOC L.2 while changes resulting from the incorporation of LCO 3.0.6 are discussed in DOC A.3.

- ITS LCO 3.0.1 includes a statement that exceptions to ITS LCO 3.0.1 are provided in LCO 3.0.2 and LCO 3.0.7. ITS LCO 3.0.2 describes the appropriate actions to be taken when ITS LCO 3.0.1 is not met. LCO 3.0.7 describes Test Exception LCOs, which are exceptions to other LCOs.

  This change is acceptable because adding the exception for LCO 3.0.2 and LCO 3.0.7 prevents a conflict within the Applicability section. This addition is needed for consistency in the ITS requirements and does not change the intent or application of the Technical Specifications. Changes resulting from the incorporation of LCO 3.0.2 are discussed in DOC A.2 while changes resulting from the incorporation of LCO 3.0.7 are discussed in DOC A.4.

These changes are designated administrative because they are editorial and result in no technical changes to the Technical Specifications.

A.3 ITS LCO 3.0.6 is added to the CTS to provide guidance regarding the appropriate ACTIONS to be taken when a single inoperability (a support system) also results in the inoperability of one or more related systems (supported system(s)). LCO 3.0.6 states "When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.10, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2." In the CTS, based on the intént and interpretation provided by the NRC over the years, there has been an ambiguous approach to the combined support/supported inoperability. Some of this history is summarized below:

- Guidance provided in the June 13, 1979, NRC memorandum from Brian K. Grimes (Assistant Director for Engineering and Projects) to Samuel E. Bryan (Assistant Director for Field Coordination) would indicate an intent/interpretation consistent with the proposed LCO 3.0.6, without the necessity of also requiring additional ACTIONS. That is, only the inoperable support system ACTIONS need be taken.

- Guidance provided by the NRC in their April 10, 1980, letter to all Licensees, regarding the definition of OPERABILITY and its impact as a support system on the remainder of the CTS, would indicate a similar philosophy of not taking ACTIONS for the inoperable supported equipment. However, in this case, additional actions (similar to the proposed Safety Function Determination Program actions) were addressed and required.

- Generic Letter 91-18 and a plain-English reading of the CTS provide an interpretation that inoperability, even as a result of a Technical Specification support system inoperability, requires all associated ACTIONS to be taken.

- Certain CTS contain ACTIONS such as "Declare the {supported system} inoperable and take the ACTIONS of {its Specification}." In many cases, the supported system would likely already be considered inoperable. The implication of this presentation is that the ACTIONS of the inoperable supported system would not have been taken without the specific direction to do so.

Considering the history of misunderstandings in this area, the BWR/4 ISTS, NUREG-1433, Rev. 3, was developed with Industry input and approval of the NRC to include LCO 3.0.6 and a new program, Specification 5.5.10, "Safety Function Determination Program (SFDP)." This change is acceptable since its function is to clarify existing ambiguities and to maintain actions within the realm of previous interpretations. This change is designated as administrative because it does not technically change the Technical Specifications.

A.4 ITS LCO 3.0.7 is added to the CTS. LCO 3.0.7 states "Special Operations LCOs in Section 3.10 allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Special Operations LCO is desired to be met but is not met, the ACTIONS of the Special Operations LCO shall be met. When a Special Operations LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications." This changes the CTS by adding specific guidance concerning the use of special test exception type LCOs.

The purpose of ITS LCO 3.0.7 is to provide guidance on the use of Special Operations LCOs. This change is acceptable because the CTS contain test exception Specifications (CTS 3.10.A and CTS 3.10.E) that allow certain LCOs to not be met for the purpose of special tests and operations. However, the CTS does not contain the equivalent of ITS LCO 3.0.7. As a result, there could be confusion regarding which LCOs are applicable during special tests and operations. LCO 3.0.7 was crafted to avoid that possible confusion. LCO 3.0.7 is consistent with the use and application of CTS test exception Specifications

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and does not provide any new restriction or allowance. This change is designated as administrative because it does not technically change the Technical Specifications.

A.5 CTS 4.0.A states "The surveillance requirements of this section shall be met. Each surveillance requirement shall be performed at the specified times except as allowed in B and C below." CTS 4.0.C states, in part, "Whenever the plant condition is such that a system or component is not required to be operable the surveillance testing associated with that system or component may be discontinued." CTS 4.0.D states "If it is discovered that a surveillance was not performed within the extended time interval allowed by 4.0.B, then the affected equipment shall be declared inoperable." ITS SR 3.0.1 states "SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits." The changes to the CTS are:

- CTS 4.0.A states, in part, "The surveillance requirements of this section shall be met." CTS 4.0.A also states, in part, "Each surveillance requirement shall be performed at the specified times except as allowed in . . . C below." CTS 4.0.C states "Whenever the plant condition is such that a system or component is not required to be operable the surveillance testing associated with that system or component may be discontinued." The first sentence of ITS SR 3.0.1 states "SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR." This changes the CTS by combining the two CTS requirements into a single cogent requirement.

  This change is acceptable because the requirements are identical. ITS SR 3.0.1 and CTS 4.0.A both state that SRs shall be met. ITS SR 3.0.1 also states when the SRs are required to be met (i.e., during the MODES or other specified conditions in the Applicability), while CTS 4.0.C states when SRs are not required to be met. This change combines the requirements of CTS 4.0.C with CTS 4.0.A (ITS SR 3.0.1) and describes the requirements in a positive way. In the ITS, certain SRs may not be required to be met in all MODES or conditions specified in the Applicability therefore, the phrase "unless otherwise stated" has been added. Changes to the Applicability of any SR will be discussed in the Discussion of Changes for the applicable ITS LCO.

- The second sentence of ITS SR 3.0.1 includes the statement, "Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO." This changes the CTS by adding the clarification "whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance."

This change is acceptable because it is consistent with the current use and application of the Technical Specifications.

- CTS 4.0.A states, in part, "Each surveillance requirement shall be performed at the specified times except as allowed in B ... below." CTS 4.0.D states "If it is discovered that a surveillance was not performed within the extended time interval allowed by 4.0.B, then the affected equipment shall be declared inoperable." The third sentence of ITS SR 3.0.1 states "Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3." This changes the CTS by replacing the CTS phrases "except as allowed in B ... below" and "within the extended time interval allowed by 4.0.B" with the ITS phrase "within the specified Frequency" and the CTS statement "then the affected equipment shall be declared inoperable" with the ITS statement "shall be failure to meet the LCO." In addition, a reference to ITS SR 3.0.3 (CTS 4.0.E) has been added. The CTS is also changed by combining CTS 4.0.A and CTS 4.0.D.

  The change associated with the replacement of the phrases "except as allowed by B ... below" and "within the extended time interval allowed by 4.0.B" is acceptable because the words "specified Frequency" imply that the allowance of CTS 3.0.B (ITS SR 3.0.2) still applies and the explicit reference to it not needed. The change associated with the replacement of the phrase "then the affected equipment shall be declared inoperable" with "shall be failure to meet the LCO" is acceptable because the intent of the CTS requirement has not changed. This change also provides the clarification "except as provided in SR 3.0.3." This change is acceptable since CTS 4.0.E (ITS SR 3.0.3) currently references CTS 4.0.B via a reference to CTS 4.0.D. Therefore this change simply places the reference in the proper location. The change associated with combining CTS 4.0.D with CTS 4.0.A is acceptable since the requirements are related to one another and their discussion in one Specification is more appropriate.

These changes are acceptable and designated administrative because they move and clarify information within the Technical Specifications.

A.6 CTS 4.0.B states, in part, "Specific time intervals between tests may be extended up to 25% of the surveillance interval." ITS SR 3.0.2 states "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications." This results in several changes to the CTS.

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- ITS SR 3.0.2 adds to the CTS "For Frequencies specified as 'once,' the above interval extension does not apply." This change is described in DOC M.2.

- ITS SR 3.0.2 adds to the CTS "If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance." This is described in DOC L.3.

- CTS 4.0.B states, in part, "Specific time intervals between tests may be extended up to 25% of the surveillance interval." ITS SR 3.0.2 states, in part, "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency." This change to the CTS is made to be consistent with the ITS terminology and to clarify the concept of the specified SR Frequency being met.

  The change is acceptable since it does not change the intent of the requirements.

- ITS SR 3.0.2 is also more specific regarding the start of the Frequency by stating "as measured from the previous performance or as measured from the time a specified condition of the Frequency is met." This direction is consistent with the current use and application of the Technical Specifications.

  This change is acceptable because the ITS presentation has the same intent as the CTS requirement.

- ITS SR 3.0.2 adds to the CTS the statement "Exceptions to this Specification are stated in the individual Specifications."

  This change is acceptable because it reflects practices used in the ITS that are not used in the CTS. Any changes to a Technical Specification, by inclusion of such an exception, will be addressed in the affected Technical Specification.

These changes are designated as administrative because they reflect presentation and usage rules of the ITS without making technical changes to the Technical Specifications.

A.7 These changes to CTS 4.0.B are provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the NRC for approval in NMC letter L-MT-04-036, from Thomas J. Palmisano (NMC) to USNRC, dated June 30, 2004. As such, these changes are administrative.

MORE RESTRICTIVE CHANGES

M.1 The CTS does not include any general LCO/ACTION guidance requirements. ITS LCO 3.0.3 is added to the CTS to provide guidance when an LCO is not met

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and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS. ITS LCO 3.0.3 states "When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in: a. MODE 2 within 7 hours; b. MODE 3 within 13 hours; and c. MODE 4 within 37 hours. Exceptions to this Specification are stated in the individual Specifications. Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required. LCO 3.0.3 is only applicable in MODES 1, 2, and 3." This changes the CTS by adding ITS LCO 3.0.3.

The purpose of ITS LCO 3.0.3 is to ensure a set of actions exists for all plant conditions when an LCO is not met. This change is acceptable since it provides the appropriate actions to take under certain conditions. These conditions are an associated Required Action and Completion Time is not met and no other Condition applies or the condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately. This Specification also delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs: a. The LCO is now met; b. A Condition exists for which the Required Actions have now been performed; or c. ACTIONS exist that do not have expired Completion Times. The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 4 when a shutdown is required during MODE 1 operation. In MODES 1, 2, and 3, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 4 and 5 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. The ITS LCO 3.0.3 Bases describes examples for this situation. This change is designated as more restrictive because explicit requirements have been included in the Technical Specifications to cover conditions not currently addressed in the CTS.

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M.2 CTS 4.0.B states, in part, "Specific time intervals between tests may be extended up to 25% of the surveillance interval." ITS SR 3.0.2 includes a similar requirement, but adds the following restriction: "For Frequencies specified as "once," the above interval extension does not apply." This changes the CTS by adding a restriction that Frequencies specified as "once" do not receive a 25% extension.

The purpose of the 1.25 extension allowance to Surveillance Frequencies is to allow for flexibility in scheduling tests. This change is acceptable because Frequencies specified as "once" are typically condition-based one-time only Surveillances in which the performance demonstrates the acceptability of the current condition and are not required to be repeated until the condition again applies. Such demonstrations should be accomplished within the specified Frequency without extension in order to avoid operation in unacceptable conditions. This change is designated as more restrictive because an allowance to extend Frequencies by 25% is eliminated from some Surveillances.

M.3 CTS 3.6.H.2 provides the actions for inoperable snubbers, and requires one of the following (a, b, or c) within 72 hours when one or more snubbers are inoperable: a) replace or restore the inoperable snubbers to OPERABLE status and perform an engineering evaluation or inspection of the supported components; b) determine through an engineering evaluation that the as-found condition of the snubber had no adverse effect on the supported components and that they would retain their structural integrity in the event of design basis seismic event; or c) declare the supported system inoperable and take the action required by the Technical Specifications for inoperability of that system. In the ITS, the actions for inoperable snubbers are incorporated into ITS LCO 3.0.8. When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and either: a) the snubbers not able to perform their associated support function(s) are associated with only one subsystem of a multiple subsystem supported system or are associated with a single subsystem supported system and are able to perform their associated support function within 72 hours; or b) the snubbers not able to perform their associated support function(s) are associated with more than one subsystem of a multiple subsystem supported system and are able to perform their associated support function within 12 hours. At the end of the specified period (i.e., 12 hours or 72 hours) snubbers must be able to perform their associated function(s), or the affected system LCO(s) shall be declared not met. This changes the CTS by requiring the risk associated with inoperable snubbers to be assessed and managed and requires the snubbers to restored to OPERABLE status in all cases, and in certain cases within a more restrictive Completion Time.

The purpose of CTS 3.6.H.2 is to provide a short time (72 hours) prior to requiring the affected systems to be declared inoperable, to either restore or replace inoperable snubbers or to perform an engineering analyses to assess whether the inoperable snubbers affect the OPERABILITY of the supported components. ITS LCO 3.0.8 requires the risk associated with inoperable required snubbers to be assessed and managed in all instances of snubber inoperability. ITS LCO 3.0.8 also requires all "required" inoperable snubbers to

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be restored to OPERABLE status within the specified Completion Times. It does not provide an explicit option to perform an engineering evaluation to assess whether the as-found condition of the snubber had no adverse effect on supported components. However, the wording of ITS LCO 3.0.8 (i.e., one or more "required" snubbers) continues to allow this evaluation to be performed. ITS LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single subsystem of a multiple subsystem supported system or to a single subsystem supported system. ITS LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable, provided only a single subsystem is affected. This 72 hour time is consistent with the CTS. However, ITS LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one subsystem of a multiple subsystem supported system, and allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. This 12 hour time is more restrictive than the CTS. The 12 hour Completion Time is acceptable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function. Furthermore, ITS LCO 3.0.8 requires that risk be assessed and managed. This risk assessment is not required in all cases in the CTS. The Bases for ITS LCO 3.0.8 provides guidance on how the risk must be assessed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of ITS LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function. This change is designated as more restrictive because inoperable snubbers must be restored to OPERABLE status under certain conditions within a more restrictive Completion Time and the risk associated with inoperable snubbers must always be assessed and managed.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.0.B states that the purpose of the 25% extension of the specified surveillance interval is "to accommodate normal test schedule." ITS SR 3.0.2 does not include this detail. This changes the CTS by moving details of the purpose of the 25% surveillance time interval extension from the CTS to the ITS Bases.

The removal of these details for meeting TS requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that the specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the CTS.

LESS RESTRICTIVE CHANGES

L.1 The CTS does not include any general LCO/ACTION guidance requirements. However, CTS 3.6.D.2 provides an explicit allowance that entry into a MODE is allowed when either a drywell floor drain sump monitoring system or the drywell particulate radioactivity monitoring system is inoperable. Thus, it is implicit that for all other Specifications, entry into a MODE or other specified condition in the Applicability of a Specification is not allowed. ITS LCO 3.0.4 is added to provide guidance when an LCO is not met and entry into a MODE or other specified condition in the Applicability is desired. ITS LCO 3.0.4 states "When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made: a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time; b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications; or c. When an allowance is stated in the individual value, parameter, or other Specification. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit." This changes the CTS by providing explicit guidance for entry into a MODE or other specified condition in the Applicability when an LCO is not met.

The purpose of LCO 3.0.4 is to provide guidance when an LCO is not met and entry into a MODE or other specified condition in the Applicability is desired. The change is acceptable because LCO 3.0.4 provides the appropriate guidance to enter the Applicability when an LCO is not met. LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c. LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited

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period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate. The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities to be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability. LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components. The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented. The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable. These systems are the High Pressure Coolant Injection System, Reactor Core Isolation Cooling System, and emergency diesel generators (ITS 3.5.1, ITS 3.5.3, and ITS 3.8.1, respectively). LCO 3.0.4.c allows entry into

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a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications that describe values and parameters. The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, and MODE 3 to MODE 4. Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specifications. Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO. This change is designated as less restrictive because entry into MODES or other specified conditions in the Applicability of a Specification might be made with an LCO not met as long as the plant is in compliance with LCO 3.0.4.

L.2 ITS LCO 3.0.5 has been added to establish allowances for restoring equipment to service. ITS LCO 3.0.5 states "Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY." This changes the CTS by adding the explicit allowance stated in LCO 3.0.5.

The purpose of LCO 3.0.5 is to establish an allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The change is acceptable since its sole purpose is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate: a. The OPERABILITY of the equipment being returned to service; or b. The OPERABILITY of other equipment. The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance. Many Technical

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Specification ACTIONS require an inoperable component to be removed from service, such as maintaining an isolation valve closed, disarming a control rod, or tripping an inoperable instrument channel. To allow the performance of Surveillance Requirements to demonstrate the OPERABILITY of the equipment being returned to service, or to demonstrate the OPERABILITY of other equipment or variables within limits, which otherwise could not be performed without returning the equipment to service, an exception to these Required Actions is necessary. ITS LCO 3.0.5 is necessary to establish an allowance that, although informally utilized in restoration of inoperable equipment, is not formally recognized in the CTS. Without this allowance, certain components could not be restored to OPERABLE status and a plant shutdown would ensue. Clearly, it is not the intent or desire that the Technical Specifications preclude the return to service of a suspected OPERABLE component to confirm its OPERABILITY. This allowance is deemed to represent a more stable, safe operation than requiring a plant shutdown to complete the restoration and confirmatory testing. This change is designated as less restrictive because LCO 3.0.5 will allow the restoration of equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS.

L.3 CTS 4.0.B states, in part, "Specific time intervals between tests may be extended up to 25% of the surveillance interval." ITS SR 3.0.2 includes a similar requirement, but adds the following: "If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance." This changes the CTS by adding an allowance that if a Required Action's Completion Time requires periodic performance on a "once per . . ." basis, the 25% Frequency extension applies to each performance after the initial performance.

This change is acceptable because the 25% Frequency extension given to provide scheduling flexibility for Surveillances is equally applicable to Required Actions that must be performed periodically. The initial performance is excluded because the first performance demonstrates the acceptability of the current condition. Such demonstrations should be accomplished within the specified Completion Time without extension in order to avoid operation in unacceptable conditions. This change is designated as less restrictive because additional time is provided to perform some periodic Required Actions.

L.4 CTS 4.0.C states "Discontinued surveillance tests shall be resumed less than one test interval before establishing plant conditions requiring operability of the associated system or component." ITS SR 3.0.4 states "Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit." This changes the CTS by allowing a discontinued Surveillance (a Surveillance discontinued due to being outside the Applicability of the LCO) to be met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as

DISCUSSION OF CHANGES
ITS SECTION 3.0, LCO AND SR APPLICABILITY

measured from the time a specified condition of the Frequency is met. This also changes the CTS by allowing a change in MODES or other specified conditions in the Applicability when a Surveillance is not current, provided the change in MODES or other specified conditions in the Applicability are allowed by LCO 3.0.4, are required to comply with ACTIONS, or are part of a shutdown of the unit.

The purpose of CTS 4.0.C is to ensure that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the plant. This change allowing use of the 25% Frequency extension allowance prior to changes in MODES or other specified conditions in the Applicability is acceptable because the 25% Frequency extension given to provide scheduling flexibility for Surveillances is equally applicable to discontinued Surveillance tests. The acceptability of a Surveillance test should not be affected by plant conditions. If the unit is operating, CTS 3.0.B (ITS SR 3.0.2) considers a Surveillance to be acceptable if the Surveillance is performed within 1.25 times the interval specified in the Frequency. The OPERABILITY of a system is normally not affected by plant conditions; therefore this change is appropriate and acceptable. The change that allows a change in MODES or other specified conditions in the Applicability when a Surveillance is not current, provided the change in MODES or other specified conditions in the Applicability are allowed by LCO 3.0.4, is acceptable because LCO 3.0.4 provides the proper guidance to enter the Applicability of an LCO when the LCO's Surveillance are not performed. Furthermore, failure to perform the Surveillance does not necessarily mean that the affected system or component is inoperable; just that it has not been demonstrated OPERABLE. The change that allows a change in MODES or other specified conditions in the Applicability when a Surveillance is not current, provided the change in MODES or other specified conditions in the Applicability is required to comply with ACTIONS or are a part of a shutdown of the unit is also acceptable. Normal shutdowns may be shutdowns required by Technical Specifications that are commenced early (e.g., prior to the absolutely required shutdown, such as day 2 of an allowed 7 day Completion Time) or shutdowns for other purposes such as refueling. Normal shutdowns would typically be performed with a full complement of OPERABLE safety systems consistent with the Bases of ITS LCO 3.0.4, which states "The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability." The addition of the allowance to perform a normal shutdown while relying on ACTIONS is appropriate because the Technical Specifications contain appropriate controls to ensure the safety of the unit in these conditions. As the unit transitions to lower MODES, less equipment is required to be OPERABLE. In addition, the Technical Specifications themselves are actually forcing the unit shutdown due to inoperability of safety system equipment, thus the shutdown should not be delayed just to perform routine, required Surveillances of other Technical Specification required equipment that is not otherwise known to be inoperable. This change is designated as less restrictive because changes in MODES or other specified conditions of the Applicability will be allowed under more conditions if a Surveillance is not current and will allow use of the 25% Frequency extension allowed under more conditions.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

DOC A.2 **LCO 3.0.1** LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, ~~and~~ LCO 3.0.7 , and LCO 3.0.8 . TSTF-372

DOC A.2 **LCO 3.0.2** Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

DOC M.1 **LCO 3.0.3** When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 2 within [7] hours; 1

b. MODE 3 within 13 hours; ~~and~~

c. MODE 4 within 37 hours. 2

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, and 3.

~~REVIEWER'S NOTE~~
~~The brackets around the time provided to reach MODE 2 allow a plant to extend the time from 7 hours to a plant specific time. Before the time can be changed, plant specific data must be provided to support the extended time.~~ 3

DOC L.1 **LCO 3.0.4** When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

CTS

LCO Applicability

DOC L.1 LCO 3.0.4 (continued)

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or ; 2

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

DOC L.2 LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

DOC A.3 LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.~~[12]~~ 10, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. 4

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

CTS

LCO Applicability

DOC A.4 LCO 3.0.7 Special Operations LCOs in Section 3.10 allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Special Operations LCOs is optional. When a Special Operations LCO is desired to be met but is not met, the ACTIONS of the Special Operations LCO shall be met. When a Special Operations LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

INSERT 1

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INSERT 1

LCO 3.0.8 When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. The snubbers not able to perform their associated support function(s) are associated with only one train/or subsystem of a multiple train/or subsystem supported system or are associated with a single train/or subsystem supported system and are able to perform their associated support function within 72 hours; or 5

b. The snubbers not able to perform their associated support function(s) are associated with more than one train/or subsystem of a multiple train/or subsystem supported system and are able to perform their associated support function within 12 hours. 5

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

CTS

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

CTS	SR	
4.0.A, 4.0.C, 4.0.D	SR 3.0.1	SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
4.0.B	SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
4.0.E	SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
4.0.C	SR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

CTS

SR Applicability

4.0.C SR 3.0.4 (continued)

This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

JUSTIFICATION FOR DEVIATIONS
ITS SECTION 3.0, LCO AND SR APPLICABILITY

1. The brackets have been removed and the proper plant specific information/value has been provided.
2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.
3. The Reviewer's Note is deleted as it is not part of the plant specific ITS.
4. Changes have been made to reflect changes in other Specifications.
5. Changes have been made for consistency with other Specifications (the term "trains" is not used).

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

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LCOs	LCO 3.0.1 through LCO 3.0.~~7~~ 8 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.10 and apply at all times, unless otherwise stated. (1)
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification ; and (2)

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

BASES

LCO 3.0.2 (continued)

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.10, "RCS Pressure and Temperature (P/T) Limits." 9 9

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/divisions of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies or

; 2

BASES

LCO 3.0.3 (continued)

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times. 3

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met. 2

b. A Condition exists for which the Required Actions have now been performed, or 2

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

BASES

LCO 3.0.3 (continued)

The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 4 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 4, or other applicable MODE, is not reduced. For example, if MODE 2 is reached in 2 hours, then the time allowed for reaching MODE 3 is the next 11 hours, because the total time for reaching MODE 3 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, and 3, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 4 and 5 because the unit is already in the most restrictive ~~C~~condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. 3

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.8, "Spent Fuel Storage Pool Water Level." LCO 3.7.8 has an Applicability of "During movement of irradiated fuel assemblies in the spent fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.8 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained to ~~by placing~~ the unit in a shutdown condition. The Required Action of LCO 3.7.8 of "Suspend movement of irradiated fuel assemblies in the spent fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications. 3

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (~~e.g.~~ i.e., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c. 3

BASES

LCO 3.0.4 (continued)

LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities to be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

BASES

LCO 3.0.4 (continued)

LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented.

The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable.

LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., [Primary Containment Air Temperature~~, Containment Pressure~~, and MCPR~~, Moderator Temperature Coefficient~~]), and may be applied to other Specifications based on NRC plant specific approval. 4

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

BASES

LCO 3.0.4 (continued)

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, and MODE 3 to MODE 4.

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 $\boxed{\text{AND}}$ LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specifications. and 3

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

a. The OPERABILITY of the equipment being returned to service or ; 2

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

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An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing. 8

BASES

LCO 3.0.5 (continued)

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the plant is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

BASES

LCO 3.0.6 (continued)

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Specification 5.5.$\boxed{12}$, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross division checks to identify a loss of safety function for those support systems that support safety systems are required. The cross division check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. A loss of safety function may exist when a support system is inoperable, and: 4

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable (EXAMPLE B 3.0.6-1),

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable (EXAMPLE B 3.0.6-2), or ; 2

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable (EXAMPLE B 3.0.6-3).

<u>EXAMPLE B 3.0.6-1</u> Division 5

If System 2 of $\boxed{\text{Train}}$ A is inoperable and System 5 of $\boxed{\text{Train}}$ B is inoperable, a loss of safety function exists in supported System 5.

<u>EXAMPLE B 3.0.6-2</u> Division 5

If System 2 of $\boxed{\text{Train}}$ A is inoperable, and System 11 of $\boxed{\text{Train}}$ B is inoperable, a loss of safety function exists in System 11 which is in turn supported by System 5.

<u>EXAMPLE B 3.0.6-3</u> Division 5

If System 2 of $\boxed{\text{Train}}$ A is inoperable, and System 1 of $\boxed{\text{Train}}$ B is inoperable, a loss of safety function exists in Systems 2, 4, 5, 8, 9, 10 and 11. 4

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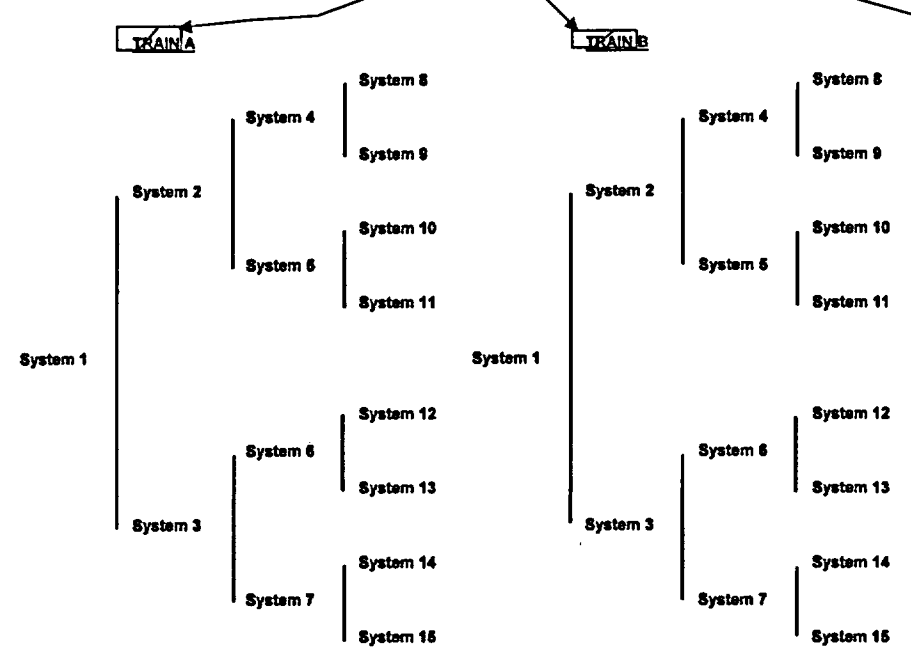
BASES

LCO 3.0.6 (continued)

If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

Division (5)

move to end of section (6)



Division (5)

Figure B 3.0-1
Configuration of ~~Trains~~ and Systems (4)

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operations ~~is~~ are being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY). (10)

BASES

LCO 3.0.6 (continued)

When loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately addresses the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system. 10

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Special Operations LCOs in Section 3.10 allow specified TS requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Special Operations LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Special Operations LCOs is optional. A special operation may be performed either under the provisions of the appropriate Special Operations LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Special Operations LCO, the requirements of the Special Operations LCO shall be followed. When a Special Operations LCO requires another LCO to be met, only the requirements of the LCO statement are required to be met regardless of that LCO's Applicability (i.e., should the requirements of this other LCO not be met, the ACTIONS of the Special Operations LCO apply, not the ACTIONS of the other LCO). However, there are instances where the Special Operations LCO ACTIONS may direct the other LCOs ACTIONS be met. The Surveillances of the other LCO are not required to be met, unless specified in the Special Operations LCO. If conditions exist such that the Applicability of any other LCO is met, all the other LCO's requirements (ACTIONS and SRs) are required to be met concurrent with the requirements of the Special Operations LCO. 3 6

NSERT from page B 3.0-10

INSERT 1 TSTF-372

TSTF-372

INSERT 1

LCO 3.0.8

LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.

If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train/or subsystem of a multiple train/or subsystem supported system or to a single train/or subsystem supported system. LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system. 8

LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train/or subsystem of a multiple train/or subsystem supported system. LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function. 8

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train/or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function. 8

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.10 and apply at all times, unless otherwise stated. 1

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified Frequency. Additionally, the definitions related to instrument testing (e.g., CHANNEL CALIBRATION) specify that these tests are performed by means of any series of sequential, overlapping, or total steps.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs ; or 2

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a Special Operations LCO are only applicable when the Special Operations LCO is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition. 7

BASES

SR 3.0.1 (continued)

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Control Rod Drive maintenance during refueling that requires scram testing at > [800 psi]. However, if other appropriate testing is satisfactorily completed and the scram time testing of SR 3.1.4.3 is satisfied, the control rod can be considered OPERABLE. This allows startup to proceed to reach [800 psi] to perform other necessary testing. 9 4

b. High pressure coolant injection (HPCI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPCI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

BASES

SR 3.0.2 (continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 3.0.2 does not apply is in the Primary Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

BASES

SR 3.0.3 (continued)

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." 8
This Regulatory Guide addresses consideration of temporary and

BASES

SR 3.0.3 (continued)

aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

A provision is included to allow entry into a MODE or other specified condition in the Applicability when an LCO is not met due to a Surveillance not being met in accordance with LCO 3.0.4.

BASES

SR 3.0.4 (continued)

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3. 3

The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, and MODE 3 to MODE 4.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO's Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

3

JUSTIFICATION FOR DEVIATIONS
ITS SECTION 3.0 BASES, LCO AND SR APPLICABILITY

1. The LCO and SR Applicability only apply to Specifications in Sections 3.1 through 3.10; they do not apply to Specifications in Chapters 4.0 and 5.0, unless specifically stated in the individual Specification. Therefore, this statement has been added for clarity.

2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

3. Typographical/grammatical error corrected.

4. The brackets have been removed and the proper plant specific information/value has been provided.

5. The Bases are changed to reflect the terminology in the definition of OPERABLE-OPERABILITY.

6. The Figure has been moved to the end of the Section, consistent with the format of the ITS.

7. The ITS SR 3.0.1 Bases allows credit to be taken for unplanned events that satisfy Surveillances. The Bases further states that this allowance also includes those SRs whose performance is normally precluded in a given MODE or other specified condition. This portion of the allowance has been deleted. As documented in Part 9900 of the NRC Inspection Manual, Technical Guidance - Licensee Technical Specifications Interpretations, and in the Bases Control Program (ITS 5.5.10), neither the Technical Specifications Bases nor Licensee generated interpretations can be used to change the Technical Specification requirements. Thus, if the Technical Specifications preclude performance of an SR in certain MODES (as is the case for some SRs in ITS Section 3.8), the Bases cannot change the Technical Specifications requirement and allow the SR to be credited for being performed in the restricted MODES, even if the performance is unplanned.

8. Changes have been made for consistency with similar discussions/terminology in the Bases.

9. Changes have been made to reflect changes in other Specifications.

10. These changes are made consistent with TSTF-482, Rev. 0, which has been approved by the USNRC for incorporation into Revision 3.1 of NUREG-1433 as documented in a letter from T. H. Boyce (NRC) to the Technical Specifications Task Force, dated 12/6/05.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS SECTION 3.0, LCO AND SR APPLICABILITY

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.1

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1433.

CTS 3/4.0 does not include any general LCO/ACTION guidance requirements. However, CTS 3.6.D.2 provides an explicit allowance that entry into a MODE is allowed when either a drywell floor drain sump monitoring system or the drywell particulate radioactivity monitoring system is inoperable. Thus, it is implicit that for all other Specifications, entry into a MODE or other specified condition in the Applicability of a Specification is not allowed. ITS LCO 3.0.4 is added to provide guidance when an LCO is not met and entry into a MODE or other specified condition in the Applicability is desired. ITS LCO 3.0.4 states "When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made: a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time; b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications; or c. When an allowance is stated in the individual value, parameter, or other Specification. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit." This changes the CTS by providing explicit guidance for entry into a MODE or other specified condition in the Applicability when an LCO is not met.

The purpose of LCO 3.0.4 is to provide guidance when an LCO is not met and entry into a MODE or other specified condition in the Applicability is desired. The change is acceptable because LCO 3.0.4 provides the appropriate guidance to enter the Applicability when an LCO is not met. LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It allows placing the unit in a MODE or other specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the LCO would not be met, in accordance with LCO 3.0.4.a, LCO 3.0.4.b, or LCO 3.0.4.c. LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate. The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities to be assessed and managed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to plan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the LCO would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability. LCO 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components. The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The LCO 3.0.4.b risk assessments do not have to be documented. The Technical Specifications allow continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the LCO, the use of the LCO 3.0.4.b allowance should be generally acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable. These systems are the High Pressure Coolant Injection System, Reactor Core Isolation Cooling System, and emergency diesel generators (ITS 3.5.1, ITS 3.5.3, and ITS 3.8.1, respectively). LCO 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the LCO not met based on a Note in the Specification which states LCO 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications that describe values and parameters. The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, and MODE 3 to MODE 4. Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specifications. Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO. This change is designated as less restrictive because entry into MODES or other specified conditions in the Applicability of a Specification might be made with an LCO not met as long as the plant is in compliance with LCO 3.0.4.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change provides explicit guidance for entry into a MODE or other specified condition in the Applicability when an LCO is not met. If the inoperability of a component or variable could increase the probability of an accident previously evaluated, the corresponding ACTIONS would not allow operation in that condition for an unlimited period of time; the risk assessment will not allow entry into the condition, and an allowance will not be provided in accordance with LCO 3.0.4. As a result, the probability of an accident previously evaluated is not significantly affected by this change. ACTIONS which allow operation for an unlimited period of time with an inoperable component or variable provide compensatory measures that protect the affected safety function, including any mitigation actions assumed in accidents previously evaluated. For example, inoperable isolation valves are closed or inoperable instrument channels are placed in trip. Since the affected safety functions continue to be protected, the mitigation functions of the component or variable continue to be performed. The risk assessment, for the purposes of LCO 3.0.4.b, must take into account all inoperable Technical Specification equipment. Therefore, entry will not be allowed if there is a loss of safety functions. Finally a Note permits the use of the provisions of LCO 3.0.4.c in LCO 3.4.5, "RCS Leakage Detection Instrumentation," and LCO 3.4.6, "RCS Specific Activity." This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is acceptable for LCO 3.4.5 as documented in the NRC safety evaluation for Technical Specification Amendment 137, dated August 21, 2003 and for LCO 3.4.6 due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions. As a result, the consequences of any accident previously

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS SECTION 3.0, LCO AND SR APPLICABILITY

evaluated are not increased significantly. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change allows entering a MODE or other specified condition in the Applicability when the allowances of LCO 3.0.4 are met. This change will not physically alter the plant (no new or different type of equipment will be installed). The change also does not require any new or revised operator actions in that operation of the unit while complying with ACTIONS is common. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change allows entering a MODE or other specified condition in the Applicability when the allowances of LCO 3.0.4 are met. This change will allow unit operation in MODES or other specified conditions in the Applicability while relying on ACTIONS that would have been previously prohibited. However, LCO 3.0.4 will only allow entry as long as the safety function is maintained. As a result, the margin of safety is not significantly affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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ITS SECTION 3.0, LCO AND SR APPLICABILITY

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.2

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1433.

ITS LCO 3.0.5 has been added to establish allowances for restoring equipment to service. ITS LCO 3.0.5 states "Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY." This changes the CTS by adding the explicit allowance stated in LCO 3.0.5.

The purpose of LCO 3.0.5 is to establish an allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The change is acceptable since its sole purpose is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate: a. The OPERABILITY of the equipment being returned to service; or b. The OPERABILITY of other equipment. The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance. Many Technical Specification ACTIONS require an inoperable component to be removed from service, such as maintaining an isolation valve closed, disarming a control rod, or tripping an inoperable instrument channel. To allow the performance of Surveillance Requirements to demonstrate the OPERABILITY of the equipment being returned to service, or to demonstrate the OPERABILITY of other equipment or variables within limits, which otherwise could not be performed without returning the equipment to service, an exception to these Required Actions is necessary. ITS LCO 3.0.5 is necessary to establish an allowance that, although informally utilized in restoration of inoperable equipment, is not formally recognized in the CTS. Without this allowance, certain components could not be restored to OPERABLE status and a plant shutdown would ensue. Clearly, it is not the intent or desire that the Technical Specifications preclude the return to service of a suspected OPERABLE component to confirm its OPERABILITY. This allowance is deemed to represent a more stable, safe operation than requiring a plant shutdown to complete the restoration and confirmatory testing. This change is designated as less restrictive because LCO 3.0.5 will allow the restoration of equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS SECTION 3.0, LCO AND SR APPLICABILITY

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change adds an allowance for restoring equipment to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. ITS LCO 3.0.5 is necessary to establish an allowance that, although informally utilized in restoration of inoperable equipment, is not formally recognized in the CTS. Without this allowance, certain components could not be restored to OPERABLE status and a plant shutdown would ensue. Clearly, it is not the intent or desire that the Technical Specifications preclude the return to service of a suspected OPERABLE component to confirm its OPERABILITY. This allowance is deemed to represent a more stable, safe operation than requiring a plant shutdown to complete the restoration and confirmatory testing. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change adds an allowance for restoring equipment to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This change will not physically alter the plant (no new or different type of equipment will be installed). Also, the change does not involve any new or revised operator actions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change adds an allowance for restoring equipment to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. The margin of safety is not affected by this change because without this allowance, certain components could not be restored to OPERABLE status and a plant shutdown would ensue. ITS LCO 3.0.5 is necessary to establish an allowance that, although informally utilized in restoration of inoperable equipment, is not formally recognized in the CTS. Without this allowance, certain components could not be restored to OPERABLE status and a plant shutdown would ensue. Clearly, it is not the intent or desire that the Technical Specifications preclude the return to service of a suspected OPERABLE component to confirm its OPERABILITY. This allowance is deemed to represent a more stable, safe operation than requiring a plant shutdown to complete the restoration and confirmatory testing. Thus, the margin of safety impact is no different than that currently exists when equipment

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ITS SECTION 3.0, LCO AND SR APPLICABILITY

is restored to service. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS SECTION 3.0, LCO AND SR APPLICABILITY

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.3

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1433.

CTS 4.0.2 states, in part, "Specific time intervals between tests may be extended up to 25% of the surveillance interval." ITS SR 3.0.2 includes a similar requirement, but adds the following: "If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance." This changes the CTS by adding an allowance that if a Required Action's Completion Time requires periodic performance on a "once per . . ." basis, the 25% Frequency extension applies to each performance after the initial performance.

This change is acceptable because the 25% Frequency extension given to provide scheduling flexibility for Surveillances is equally applicable to Required Actions that must be performed periodically. The initial performance is excluded because the first performance demonstrates the acceptability of the current condition. Such demonstrations should be accomplished within the specified Completion Time without extension in order to avoid operation in unacceptable conditions. This change is designated as less restrictive because additional time is provided to perform some periodic Required Actions.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change allows the Completion Time for periodic actions to be extended by 25%. This change does not affect the probability of an accident. The length of time between performance of Required Actions is not an initiator to any accident previously evaluated. The consequences of any accident previously evaluated are the same during the Completion Time or during any extension of the Completion Time. As a result, the consequences of any accident previously evaluated are not increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS SECTION 3.0, LCO AND SR APPLICABILITY

The proposed change allows the Completion Time for periodic actions to be extended by 25%. This change will not physically alter the plant (no new or different type of equipment will be installed). Also, the change does not involve any new or revised operator actions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

The proposed change allows the Completion Time for periodic actions to be extended by 25%. The 25% extension allowance is provided for scheduling convenience and is not expected to have a significant effect on the average time between Required Actions. As a result, the Required Actions will continue to provide appropriate compensatory measures for the subject Condition. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS SECTION 3.0, LCO AND SR APPLICABILITY

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.4

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1433.

CTS 4.0.C states "Discontinued surveillance tests shall be resumed less than one test interval before establishing plant conditions requiring operability of the associated system or component." ITS SR 3.0.4 states "Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit." This changes the CTS by allowing a discontinued Surveillance (a Surveillance discontinued due to being outside the Applicability of the LCO) to be met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. This also changes the CTS by allowing a change in MODES or other specified conditions in the Applicability when a Surveillance is not current, provided the change in MODES or other specified conditions in the Applicability are allowed by LCO 3.0.4, are required to comply with ACTIONS, or are part of a shutdown of the unit.

The purpose of CTS 4.0.C is to ensure that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the plant. This change allowing use of the 25% Frequency extension allowance prior to changes in MODES or other specified conditions in the Applicability is acceptable because the 25% Frequency extension given to provide scheduling flexibility for Surveillances is equally applicable to discontinued Surveillance tests. The acceptability of a Surveillance test should not be affected by plant conditions. If the unit is operating, CTS 3.0.B (ITS SR 3.0.2) considers a Surveillance to be acceptable if the Surveillance is performed within 1.25 times the interval specified in the Frequency. The OPERABILITY of a system is normally not affected by plant conditions; therefore this change is appropriate and acceptable. The change that allows a change in MODES or other specified conditions in the Applicability when a Surveillance is not current, provided the change in MODES or other specified conditions in the Applicability are allowed by LCO 3.0.4, is acceptable because LCO 3.0.4 provides the proper guidance to enter the Applicability of an LCO when the LCO's Surveillance are not performed. Furthermore, failure to perform the Surveillance does not necessarily mean that the affected system or component is inoperable; just that it has not been demonstrated OPERABLE. The change that allows a change in MODES or other specified conditions in the Applicability when a Surveillance is not current, provided the change in MODES or other specified conditions in the Applicability is required to comply with ACTIONS or are a part of a

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ITS SECTION 3.0, LCO AND SR APPLICABILITY

shutdown of the unit is also acceptable. Normal shutdowns may be shutdowns required by Technical Specifications that are commenced early (e.g., prior to the absolutely required shutdown, such as day 2 of an allowed 7 day Completion Time) or shutdowns for other purposes such as refueling. Normal shutdowns would typically be performed with a full complement of OPERABLE safety systems consistent with the Bases of ITS LCO 3.0.4, which states "The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability." The addition of the allowance to perform a normal shutdown while relying on ACTIONS is appropriate because the Technical Specifications contain appropriate controls to ensure the safety of the unit in these conditions. As the unit transitions to lower MODES, less equipment is required to be OPERABLE. In addition, the Technical Specifications themselves are actually forcing the unit shutdown due to inoperability of safety system equipment, thus the shutdown should not be delayed just to perform routine, required Surveillances of other Technical Specification required equipment that is not otherwise known to be inoperable. This change is designated as less restrictive because changes in MODES or other specified conditions of the Applicability will be allowed under more conditions if a Surveillance is not current and will allow use of the 25% Frequency extension allowed under more conditions.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change will allow a discontinued Surveillance (a Surveillance discontinued due to being outside the Applicability of the LCO) to be met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. This change also changes the CTS by allowing a change in MODES or other specified conditions in the Applicability when a Surveillance is not current, provided the change in MODES or other specified conditions in the Applicability are allowed by LCO 3.0.4, are required to comply with ACTIONS, or are part of a shutdown of the unit. Failure to perform the Surveillance does not necessarily mean that the affected system or component is inoperable; just that it has not been demonstrated OPERABLE. The length of time between performance of Surveillances is not an initiator to any accident previously evaluated. The consequences of any accident previously evaluated are the same during the normal Surveillance interval not being met or during any extension of the Surveillance interval. This change will allow unit operation in MODES or other specified conditions in the Applicability while relying on ACTIONS that would have been previously prohibited. However, LCO 3.0.4 will only allow entry if the associated ACTIONS to be entered permit continued operation for an unlimited period of time, a risk evaluation is performed prior to entry into the MODE, or when specific analysis has been previously approved allowing entry. This change also allows a change in MODES or other specified conditions in the Applicability when a Surveillance is not current provided entry is required to comply with ACTIONS or are part of a shutdown of the unit. Normal shutdowns would typically be performed with a full complement of OPERABLE safety systems consistent with the Bases of ITS LCO 3.0.4, which states "The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability." The addition of the allowance to perform a normal shutdown while relying on ACTIONS is appropriate because the Technical Specifications contain appropriate controls to ensure the safety of the unit in these conditions. These allowances are not considered to increase the probability of an accident previously evaluated or significantly increase the consequences of an accident previously evaluated since the failure to perform the Surveillance does not necessarily mean that the affected system or component is inoperable; just that it has not been demonstrated OPERABLE. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No.

The proposed change will allow a discontinued Surveillance (a Surveillance discontinued due to being outside the Applicability of the LCO) to be met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. This change also changes the CTS by allowing a change in MODES or other specified conditions in the Applicability when a Surveillance is not current, provided the change in MODES or other specified conditions in the Applicability are allowed by LCO 3.0.4, are required to comply with ACTIONS, or are part of a shutdown of the unit. These proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

The extended Frequency for discontinued Surveillances tests will ensure the equipment is OPERABLE prior to entry into the proposed Applicability therefore this change does not result in a significant reduction in the margin of safety. The acceptability of a Surveillance test should not be affected by plant conditions. If the unit is operating CTS 3.0.B (ITS SR 3.0.1) considers a Surveillance to be acceptable if the Surveillance is performed within 1.25 times the interval specified in the Frequency. CTS 3.0.C does not allow the 25% extension of the Frequency if the plant is outside the Applicability of the Specification and the Surveillance has been discontinued. The OPERABILITY of a system is normally not affected

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS SECTION 3.0, LCO AND SR APPLICABILITY

by plant conditions; therefore this change is appropriate and acceptable. Thus, there is confidence that the equipment can perform its assumed safety function. This change also changes the CTS by allowing a change in MODES or other specified conditions in the Applicability when a Surveillance is not current, provided the change in MODES or other specified conditions in the Applicability are allowed by LCO 3.0.4, are required to comply with ACTIONS, or are part of a shutdown of the unit. This change will allow unit operation in MODES or other specified conditions in the Applicability while relying on ACTIONS that would have been previously prohibited. However, LCO 3.0.4 will only allow entry if the associated ACTIONS to be entered permit continued operation for an unlimited period of time, a risk evaluation is performed prior to entry into the MODE, or when specific analysis has been previously approved allowing entry. This change also allows a change in MODES or other specified conditions in the Applicability when a Surveillance is not current provided entry is required to comply with ACTIONS or are part of a shutdown of the unit. Normal shutdowns would typically be performed with a full complement of OPERABLE safety systems consistent with the Bases of ITS LCO 3.0.4, which states "The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability." The addition of the allowance to perform a normal shutdown while relying on ACTIONS is appropriate because the Technical Specifications contain appropriate controls to ensure the safety of the unit in these conditions. These controls are considered adequate to maintain the margin of safety. As a result, the margin of safety is not significantly affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

IMPROVED TECHNICAL SPECIFICATIONS

MONTICELLO NUCLEAR GENERATING PLANT

VOLUMES 6 & 7
REVISION 1

6. ITS Section 3.1, - Reactivity Control Systems
7. ITS Section 3.2, - Power Distribution Limits

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Summary of Changes
ITS Section 3.1

Change Description	Affected Pages
The changes described in the NMC response to Question 200510141334 have been made. Changes are made to be consistent with TSTF-439, Rev. 2 (Eliminate Second Completion Times Limiting Time From Discovery of Failure to Meet an LCO).	Pages 167, 169, 175, 181, 189, 192, 193, and 200 of 231
The changes described in the NMC response to Question 200601201446 have been made. Minor typographical error in the NUREG (ITS Markup) has been corrected.	Pages 131 and 133 of 231

ATTACHMENT 1

VOLUME 6

MONTICELLO IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.1 REACTIVITY CONTROL SYSTEMS

Revision 1

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS 3.1.1, SHUTDOWN MARGIN (SDM)

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS | ITS

Attachment 1, Volume 6, Rev. 1, Page 5 of 231

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3.3 CONTROL ROD SYSTEM	4.3 CONTROL ROD SYSTEM
~~Applicability:~~ ~~Applies to the operational status of the control rod system.~~ ~~Objective:~~ ~~To assure the ability of the control rod system to control reactivity.~~	~~Applicability:~~ ~~Applies to the surveillance requirements of the control rod system.~~ ~~Objective:~~ ~~To verify the ability of the control rod system to control reactivity.~~
~~Specification:~~ ~~A. Reactivity Limitations~~	~~Specification:~~ ~~A. Reactivity Limitations~~
3.1.1 — 1. Reactivity ~~margin - core loading~~ [SHUTDOWN MARGIN (SDM)] (LA.1)	1. Reactivity ~~margin~~ - core loading
LCO 3.1.1 — [SDM shall be:] (A.1) ~~The core loading shall be limited to that~~ which can be made subcritical ~~in the most reactive condition during the operating cycle~~ with the strongest operable control rod in its full-out position and all other operable rods fully inserted. { See ITS 1.0 }	SR 3.1.1.1 — ~~Sufficient control rods shall be withdrawn~~ [Verify SDM to be within limits.] (LA.2) ~~following a refueling outage when core alterations were performed~~ [Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement] (M.3) (LA.2) ~~to demonstrate~~ LCO 3.1.1 — ~~with a margin of 0.25 per cent Δk~~ (M.1) that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted. [Add proposed SR 3.1.1.1 first Frequency] (M.4) { See ITS 1.0 }
[APPLICABILITY: MODES 1, 2, 3, 4, and 5] (M.2)	

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A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
F. Scram Discharge Volume 1. During reactor operation, the scram discharge volume vent and drain valves shall be operable, except as specified below. 2. If any scam discharge volume vent or drain valve is made or found inoperable, the integrity of the scram discharge volume shall be maintained by either: a. Verifying daily, for a period not to exceed 7 days, the operability of the redundant valve(s), or b. Maintaining the inoperable valve(s), or the associated redundant valve(s), in the closed position. Periodically the inoperable and the redundant valve(s) may both be in the open position to allow draining the scram discharge volume. If a or b above cannot be met, at least all but one operable control rods (not including rods removed per specification 3.10.E or inoperable rods allowed by 3.3.A.2) shall be fully inserted within ten hours.	F. Scram Discharge Volume The scram discharge volume vent and drain valves shall be cycled quarterly. Once per operating cycle verify the scram discharge volume vent and drain valves close within 30 seconds after receipt of a reactor scram signal and open when the scram is reset.

See ITS 3.1.8

A.2

ACTIONS A, B, C, and D

1.

G. Required Action

If Specifications 3.3.A through D above are not met, ~~an orderly shutdown shall be initiated and have reactor in the cold shutdown condition within 24 hours~~.

(except when the reactor mode switch is in the Refuel position) A.2

Add proposed ACTIONS A, B, C, and D M.5

INSERT A A.2

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A.1

ITS

A.2

INSERT A

ACTION E

2. If Specification 3.3.A is not met when the reactor mode switch is in the Refuel position, immediately suspend core alterations except for fuel assembly removal and control rod insertion and immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Add proposed Required Actions D.3 and D.4

A.3

M.6

DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 These changes to CTS 3.3.G are provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the USNRC for approval in NMC letter L-MT-05-013, from Thomas J. Palmisano (NMC) to USNRC, dated April 12, 2005. As such, these changes are administrative.

A.3 CTS 3.3.G.2 requires the immediate suspension of core alterations except for "fuel assembly removal" and to "immediately initiate action to fully insert all insertable control rods in core cell containing one or more fuel assemblies" if CTS 3.3.A is not met when the reactor mode switch is in the Refuel position. ITS 3.1.1 ACTION E covers the condition for SDM not met in MODE 5, and in part, requires the immediate suspension of CORE ALTERATIONS except for "control rod insertion and fuel assembly removal" and requires the immediate initiation of action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. This changes the CTS by clarifying that CORE ALTERATIONS that involve "the insertion of control rods" are also excepted.

The purpose of CTS 3.3.G.2 is to immediately stop all core alterations that can reduce shutdown margin. The CTS definition (CTS 1.0.A) of "Alteration of the Reactor Core" does not include normal operating functions such as control rod movement using the normal drive mechanism. In ITS 1.1, the "CORE ALTERATIONS" definition includes the movement of control rods as long as the associated core cell contains one or more fuel assemblies. This change is acceptable because CTS 3.3.G.2 specifically requires action to fully insert all insertable control rods in core cells containing one more fuel assemblies. Therefore, the addition of the exception is considered administrative. This change is designated as administrative because it does not represent a technical change to the Technical Specifications.

MORE RESTRICTIVE CHANGES

M.1 CTS 4.3.A.1 states, in part, reactivity margin of "0.25 per cent Δk" is required. ITS LCO 3.1.1 states SDM shall be: a. $\geq 0.38\% \Delta k/k$, with the highest worth control rod analytically determined; or b. $\geq 0.28\% \Delta k/k$, with the highest worth control rod determined by test. This changes the CTS by replacing the existing SDM limit with two new limits.

The purpose of ITS LCO 3.1.1 is to allow flexibility in the determination of SDM. This change is acceptable because the LCO requirements continue to ensure

DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

that the reactor core is maintained consistent with the safety analyses. ITS LCO 3.1.1 provides a SDM of ≥ 0.38% Δk/k, with the highest worth control rod analytically determined or a SDM limit of ≥ 0.28% Δk/k, with the highest worth control rod determined by test. The current limit of ≥ 0.25% Δk/k does not specify how the strongest control rod is determined. This change is acceptable because for the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10% Δk/k) must be added to the SDM limit for uncertainties in the calculation. The SDM may be demonstrated during an in sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local control rod tests, where the highest worth control rod is determined by testing. The proposed allowances are consistent with the ISTS and the additional margin is considered sufficient based on the uncertainties observed in the calculation methodology. This change is designated as more restrictive since the new limits will require additional SDM in order to satisfy the Specification.

M.2 CTS 3.3.A.1 states, in part, that core loading shall be limited to that which can be made subcritical in the most reactive condition during the operating cycle. CTS 4.3.A.1 states, in part, that a test shall be performed to demonstrate that the core can be made subcritical at any time in the subsequent fuel cycle. CTS 3.3.G.1 requires the unit to be in cold shutdown (MODE 4) within 24 hours if CTS 3.3.A.1 is not met. CTS 3.3.G.2 provides Actions for when the reactor mode switch is in the Refuel position (i.e., MODE 5 in the ITS). ITS LCO 3.1.1 requires SDM to be met during MODES 1, 2, 3, 4, and 5. This changes the CTS by changing the Applicability from MODE 1, 2, and 3 (based on the shutdown requirement of CTS 3.3.G.1) and MODE 5 (based on the reactor mode switch position requirement of CTS 3.3.G.2) to MODES 1, 2, 3, 4, and 5. Changes to the requirements of CTS 3.3.G.1 are discussed in DOC M.5 and changes to the requirements of CTS 3.3.G.2 are discussed in DOCs A.3 and M.6.

The purpose of the ITS 3.1.1 Applicability is to ensure SDM is met whenever fuel is in the reactor core. The change is acceptable because the safety analyses assume that SDM is met whenever fuel is in the reactor core. In MODES 1 and 2, SDM must be provided because subcriticality with the highest worth control rod withdrawn is assumed in the control rod drop accident analysis and other accident and transient analyses. In MODES 3 and 4, SDM is required to ensure the reactor will be held subcritical with margin for a single withdrawn control rod. SDM is required in MODE 5 to prevent an open vessel, inadvertent criticality during the withdrawal of a single control rod from a core cell containing one or more fuel assemblies or a fuel assembly insertion error. This change is designated as more restrictive because it increases the conditions for when the Specification is required to be met.

M.3 CTS 4.3.A.1 states, in part, the reactivity margin demonstration shall be performed "following a refueling outage when core alterations were performed." ITS SR 3.1.1.1 states, verify SDM to be within limits at a Frequency of "Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement." This changes the CTS by stating a finite time to complete the Surveillance (once within 4 hours after criticality) and requiring the Surveillance to be performed following fuel movement within the reactor

DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

pressure vessel or control rod replacement in lieu of following a "refueling outage" when core alterations were performed.

The purpose of CTS 4.3.A.1 is to ensure there is sufficient reactivity margin designed in the reactor core and that this is demonstrated after a refueling outage after core alterations are made. The proposed Surveillance Frequency in ITS SR 3.1.1.1 states that a SDM demonstration must be performed "Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement" instead of the current requirement to perform the demonstration "following a refueling outage when core alterations were performed." Therefore, this change effectively places a finite time limit on completing the Surveillance. In addition, the current Surveillance is only required after a refueling outage. The intent of this portion of the Surveillance Frequency is to verify the core reactivity after in-vessel operations that could have altered the core reactivity. During refueling outages, core reactivity is normally significantly altered. However, conditions could arise mid-cycle that require replacing a fuel assembly or control rod, and the CTS would not require this Surveillance to be performed during the subsequent reactor startup. This mid-cycle replacement has the potential of altering core reactivity. The ITS words cover both planned refueling outages and other outages where CORE ALTERATIONS may occur, thus this change is considered acceptable. This change is acceptable since the proposed Frequency ensures SDM is within limits shortly after any fuel movement within the reactor pressure vessel or any control rod replacements have been made. The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification. This change is designated as more restrictive because a Surveillance will be performed under more conditions and with a finite time limit for completion under the ITS than under the CTS.

M.4 ITS SR 3.1.1.1 requires verification of SDM "Prior to each in vessel fuel movement during fuel loading sequence." Currently, the CTS does not require a SDM verification at this Frequency. This changes the CTS by adding a new Surveillance Frequency for the SDM verification.

The purpose of the new Surveillance Frequency in ITS SR 3.1.1.1 (first Frequency) is to ensure SDM is met during the fuel loading sequence. This change adds a requirement to ensure SDM is met "Prior to each in vessel fuel movement during fuel loading sequence." This change is acceptable because the new Surveillance Frequency in ITS SR 3.1.1.1 will ensure the reactor core will not go critical during a fuel loading sequence. During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided

DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM. This change is designated as more restrictive because it adds the requirement to verify SDM during a fuel loading sequence.

M.5 CTS 3.3.G.1 requires the unit to be in cold shutdown (MODE 4) within 24 hours if CTS 3.3.A.1 is not met. ITS 3.1.1 specifies specific ACTIONS for each MODE (MODE 1, 2, 3, 4, and 5). ITS 3.1.1 ACTION A covers the condition for SDM not met in MODES 1 or 2, and requires the restoration of SDM to within limits within 6 hours. If this is not met, ITS 3.1.1 ACTION B requires the unit to be in MODE 3 in 12 hours. ITS 3.1.1 ACTION C covers the condition for SDM not met in MODE 3, and requires immediate initiation of action to fully insert all insertable control rods. ITS 3.1.1 ACTION D covers the condition for SDM not met in MODE 4, and requires immediate initiation of action to fully insert all insertable control rods, and within 1 hour, to restore secondary containment to OPERABLE status, to restore one standby gas treatment (SGT) subsystem to OPERABLE status, and to restore isolation capability in each required secondary containment penetration flow path not isolated. This changes the CTS by specifying explicit compensatory actions for MODES 1, 2, 3, and 4 in lieu of a single common action for these MODES.

The purpose of the ITS 3.1.1 ACTIONS are to ensure the appropriate compensatory actions are taken when SDM is not met. CTS 3.3.G.1 requires the unit to be in cold shutdown (MODE 4) within 24 hours when the SDM requirements are not met. In MODES 1 and 2, the ITS will allow 6 hours to restore the SDM to within limits. If this can not be met, the unit must be in MODE 3 (hot shutdown) within the next 12 hours. This portion of the change is acceptable since it reduces the time the unit must be at a specified condition (from 24 hours to cold shutdown to 18 hours to MODE 3) and places the unit in condition where the core reactivity is reduced. If the unit is brought to MODE 3 there is no requirement to go to MODE 4 (cold shutdown) since in this condition the reactivity of the core may effectively increase due to the reduction in reactor coolant temperature. Therefore, the ITS 3.1.1 compensatory action in MODE 3 (ITS 3.1.1 ACTION C) is acceptable since it will help to reduce the reactivity conditions of the core by requiring the immediate initiation of action to insert all insertable control rods. Although there is no requirement to achieve MODE 4 conditions, the proposed action to stay in MODE 3 is acceptable and appropriate considering the behavior of the reactor core. In MODE 4, the ITS compensatory actions continue to require the reduction of the core reactivity and to help minimize any consequences of an event if an event should occur during the time period when SDM is not met. The compensatory actions proposed for MODE 4 are considered appropriate and acceptable. This change is designated as more restrictive because it adds compensatory actions and reduces the time limit in which the unit must be in a specified condition.

M.6 CTS 3.3.G.2 requires the immediate suspension of core alterations except for "fuel assembly removal" and to "immediately initiate action to fully insert all insertable control rods in core cell containing one or more fuel assemblies" if CTS 3.3.A.1 is not met when the reactor mode switch is in the Refuel position. ITS 3.1.1 ACTION E covers the condition for SDM not met in MODE 5, and requires the immediate suspension of CORE ALTERATIONS except for control

DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

rod insertion and fuel assembly removal, immediate initiation of action to fully insert all insertable control rods in core cells containing one or more fuel assemblies, and to initiate action within 1 hour to restore secondary containment to OPERABLE status, restore one standby gas treatment (SGT) subsystem to OPERABLE status, and restore isolation capability in each required secondary containment penetration flow path not isolated. This changes the CTS by adding the explicit compensatory actions associated with the secondary containment functions.

The purpose of CTS 3.3.G.2 is to immediately stop all core alterations that can reduce shutdown margin. Actions have been added that require the restoration of the secondary containment, one SGT subsystem, and the isolation capability in each required secondary containment penetration flow path not isolated. These actions are provided for the control of potential radioactive release. In MODE 5, the ITS compensatory actions continue to require the reduction of the core reactivity and to help minimize any consequences of an event if an event should occur during the time period when SDM is not met. The compensatory actions proposed for MODE 5 are considered appropriate and acceptable. This change is designated as more restrictive because it adds compensatory actions.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.3.A.1 states, in part, that the core loading shall be limited to that which can be made subcritical "in the most reactive condition during the operating cycle." ITS LCO 3.1.1 requires SDM to be met. This changes the CTS by relocating the details that the core loading shall be limited to that which can be made subcritical "in the most reactive condition during the operating cycle" to the ITS Bases in the form of a discussion about how core reactivity varies during the fuel cycle and that the SDM verification should consider this behavior.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that the SDM shall be within limits. This is required all times during the operating cycle, including the most reactive condition during the operating cycle. The details of how SDM is calculated does not need to appear in the Specification in order for the requirement to apply. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated

DISCUSSION OF CHANGES
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the CTS.

LA.2 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.3.A.1 states, in part, "Sufficient control rods shall be withdrawn . . . to demonstrate" reactivity margin is within the specified limit. ITS SR 3.1.1.1 states "Verify SDM to be within limits," but does not provide similar details of how to perform the verification. This changes the CTS by relocating the test method "Sufficient control rods shall be withdrawn ... to demonstrate" reactivity margin to the ITS Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that the SDM shall be within limits and to verify the SDM limits are met. The details of how SDM is performed does not need to be stated in the Specification in order for the requirement to apply. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the CTS.

<u>LESS RESTRICTIVE CHANGES</u>

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be:

3.3.A.1, 4.3.A.1 a. ≥ [0.38] % Δk/k, with the highest worth control rod analytically determined or ; 1 2

b. ≥ [0.28] % Δk/k, with the highest worth control rod determined by test. 1

3.3.A.1 APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.3.G.1	A. SDM not within limits in MODE 1 or 2.	A.1 Restore SDM to within limits.	6 hours	
3.3.G.1	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours	
3.3.G.1	C. SDM not within limits in MODE 3.	C.1 Initiate action to fully insert all insertable control rods.	Immediately	
3.3.G.1	D. SDM not within limits in MODE 4.	D.1 Initiate action to fully insert all insertable control rods. AND D.2 Initiate action to restore [secondary] containment to OPERABLE status. AND	Immediately 1 hour	1

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION		COMPLETION TIME
3.3.G.1		D.3	Initiate action to restore one standby gas treatment (SGT) subsystem to OPERABLE status.	1 hour
		AND		
		D.4	Initiate action to restore isolation capability in each required [secondary] containment penetration flow path not isolated.	1 hour
3.3.G.2	E. SDM not within limits in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal.	Immediately
		AND		
		E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
		AND		

1

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION		COMPLETION TIME	
3.3.G.2		E.3	Initiate action to restore [secondary] containment to OPERABLE status.	1 hour	1
		AND			
		E.4	Initiate action to restore one SGT subsystem to OPERABLE status.	1 hour	
		AND			
		E.5	Initiate action to restore isolation capability in each required [secondary] containment penetration flow path not isolated.	1 hour	1

CTS

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY
4.3.A.1	SR 3.1.1.1	Verify SDM to be within limits.	Prior to each in vessel fuel movement during fuel loading sequence AND Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. This punctuation correction has been made consistent with the Writer's Guide for the Standard Technical Specifications, NEI 01-03, Section 5.1.3.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure: INSERT 1 1

a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Events,

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

USAR, Section 3.3.3.3

These requirements are satisfied by the control rods, as described in GDC 26 (Ref. 1), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions. 1

APPLICABLE SAFETY ANALYSES

INSERT 2

The control rod drop accident (CRDA) analysis (Refs. 2 and 3) assumes the core is subcritical with the highest worth control rod withdrawn. 1 Typically, the first control rod withdrawn has a very high reactivity worth and, should the core be critical during the withdrawal of the first control rod, the consequences of a CRDA could exceed the fuel damage limits for a CRDA (see Bases for LCO 3.1.6, "Rod Pattern Control"). Also, SDM is assumed as an initial condition for the control rod removal error during refueling (Ref. 4) and fuel assembly insertion error during refueling (Ref. 5) accidents. The analysis of these reactivity insertion events assumes the refueling interlocks are OPERABLE when the reactor is in the refueling mode of operation. 1 These interlocks prevent the withdrawal of more than one control rod from the core during refueling. (Special consideration and requirements for multiple control rod withdrawal during refueling are covered in Special Operations LCO 3.10.6, "Multiple Control Rod Withdrawal - Refueling.") The analysis assumes this condition is acceptable since the core will be shut down with the highest worth control rod withdrawn, if adequate SDM has been demonstrated. 1

rely on adequate SDM and proper operation of

, thereby positive ing

Prevention or mitigation of reactivity insertion events is necessary to limit energy deposition in the fuel to prevent significant fuel damage, which could result in undue release of radioactivity. 2 Adequate SDM ensures inadvertent criticalities and potential CRDAs involving high worth control rods (namely the first control rod withdrawn) will not cause significant fuel damage. 1

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

1

<u>INSERT 1</u>

a. The reactor core is designed so that control rod action, with the maximum worth control rod fully withdrawn and unavailable for use, is capable of bringing the reactor core subcritical and maintaining it so from any power level in the operating cycle; and

b. The reactor core and associated systems are designed to accommodate unit operational transients or maneuvers which might be expected without compromising safety and without fuel damage.

1

<u>INSERT 2</u>

Having sufficient SDM assures that the reactor will become and remain subcritical after all design basis accidents and transients.

BASES

LCO

The specified SDM limit accounts for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically or by measurement. This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement. When SDM is demonstrated by calculations not associated with a test (e.g., to confirm SDM during the fuel loading sequence), additional margin is included to account for uncertainties in the calculation. To ensure adequate SDM during the ~~design~~ process, a design margin is included to account for uncertainties in the design calculations (Ref. 6). 4 2 1

APPLICABILITY

In MODES 1 and 2, SDM must be provided because subcriticality with the highest worth control rod withdrawn is assumed in the CRDA analysis (Ref. 2). INSERT 3 In MODES 3 and 4, SDM is required to ensure the reactor will be held subcritical with margin for a single withdrawn control rod. SDM is required in MODE 5 to prevent an open vessel, inadvertent criticality during the withdrawal of a single control rod from a core cell containing one or more fuel assemblies [or a fuel assembly insertion error (Ref. 5)]. 1 3

ACTIONS

A.1

With SDM not within the limits of the LCO in MODE 1 or 2, SDM must be restored within 6 hours. Failure to meet the specified SDM may be caused by a control rod that cannot be inserted. The allowed Completion Time of 6 hours is acceptable, considering that the reactor can still be shut down, assuming no failures of additional control rods to insert, and the low probability of an event occurring during this interval.

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 in 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1

With SDM not within limits in MODE 3, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core.

1

INSERT 3

and other design basis accidents and transients

BASES

ACTIONS (continued)

D.1, D.2, D.3, and D.4

With SDM not within limits in MODE 4, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; is available at least one Standby Gas Treatment (SGT) subsystem is OPERABLE; and secondary containment isolation capability (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability) in each associated penetration flow path not isolated INSERT 4 that is assumed to be isolated to mitigate radioactivity releases. This may INSERT 5 be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

2 3 2 2

E.1, E.2, E.3, E.4, and E.5

With SDM not within limits in MODE 5, the operator must immediately suspend CORE ALTERATIONS that could reduce SDM (e.g., insertion of fuel in the core or the withdrawal of control rods). Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and are therefore excluded from the suspended actions.

Action must also be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and therefore do not have to be inserted.

2

INSERT 4

These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

2

INSERT 5

(ensuring components are OPERABLE)

BASES

ACTIONS (continued)

2

Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one SGT subsystem is OPERABLE; and secondary containment isolation capability (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability) in each associated penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances as needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Action must continue until all required components are OPERABLE.

is available

3

2

INSERT 6

INSERT 7

2

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

verified

This can be accomplished by a test, an evaluation, or a combination of the two.

4

Adequate SDM must be demonstrated to ensure that the reactor can be made subcritical from any initial operating condition. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, control rod replacement, or shuffling within the reactor pressure vessel. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 7). For the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10% Δk/k) must be added to the SDM limit as specified in the COLR to account for uncertainties in the calculation.

2

2

of core reactivity 2

5

1

4

of 0.28% Δk/k

The SDM may be demonstrated during an in-sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local criticals, where the highest worth control rod is determined by testing.

5

1

control rod tests

2

INSERT 6

These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

2

INSERT 7

(ensuring components are OPERABLE)

BASES

SURVEILLANCE REQUIREMENTS (continued)

control rod

Local ~~critical~~ tests require the withdrawal of out of sequence control rods. This testing ~~would~~ could therefore require bypassing of the rod worth minimizer to allow the out of sequence withdrawal, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing - Operating"). 1 4

The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During MODES 3 and 4, analytical calculation of SDM may be used to assure the requirements of SR 3.1.1.1 are met.

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM. 2

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 26~~ USAR, Section 3.3.3.3. 1
2. ~~F~~ U SAR, Section ~~[15.1.38]~~ 14.7.1. 1 3
3. NEDE-24011-P-A~~-9-US~~, "General Electric Standard Application for Reactor Fuel," Supplement for United States, Section S.2.2.3.1, ~~September 1988~~ (revision specified in Specification 5.6.3). 1 1
~~4. FSAR, Section [15.1.13].~~
~~5. FSAR, Section [15.1.14].~~ 3
4 ~~6~~. ~~F~~ U SAR, Section ~~[4.3.2.4.1]~~ 14A.3.1. 1 3 1
5 ~~7~~. NEDE-24011-P-A~~-9~~, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, ~~September 1988~~ (revision specified in Specification 5.6.3). 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.1 BASES, SHUTDOWN MARGIN (SDM)

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.
2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.
3. The brackets have been removed and the proper plant specific information has been provided.
4. The Bases have been changed to reflect the Specification.
5. Typographical/grammatical error corrected.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.1, SHUTDOWN MARGIN (SDM)

There are no specific NSHC discussions for this Specification.

ATTACHMENT 2

ITS 3.1.2, Reactivity Anomalies

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS		ITS	
	3.0 LIMITING CONDITIONS FOR OPERATION		**4.0 SURVEILLANCE REQUIREMENTS**
3.1.2	E. Reactivity Anomalies		E. Reactivity Anomalies
SR 3.1.2.1 LCO 3.1.2 ACTION A Applicability	~~At a specific steady-state base condition~~ of the reactor actual control rod inventory will be periodically compared to a ~~normalized computed~~ prediction of the inventory. If the difference exceeds one per cent, delta k, ~~reactor power operation~~ shall not be permitted until the cause has been evaluated and appropriate corrective action has been completed.	SR 3.1.2.1	~~During the startup test program and~~ ~~at each~~ startup ~~following refueling outages,~~ the actual rod inventory shall be compared to a ~~normalized computed~~ prediction of the inventory. ~~These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle.~~ ~~At specific power operating conditions,~~ the actual rod configuration will be compared to the configuration expected ~~based upon appropriately corrected past data.~~ This comparison will be made at least every ~~equivalent full power month.~~

LA.1 M.3 L.1

Add proposed ACTIONS A and B

A.2 M.1

Once within 24 hours after reaching equilibrium conditions following

after fuel movement within the reactor pressure vessel or control rod replacement

M.2 LA.1 L.2 LA.1

1000 MWD/T thereafter during operations in MODE 1

L.2

3.3/4.3

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DISCUSSION OF CHANGES
ITS 3.1.2, REACTIVITY ANOMALIES

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 4.3.E states, in part, the reactivity anomaly Surveillance must be performed "During the startup test program." ITS SR 3.1.2.1 does not include this requirement. This changes the CTS by deleting the requirement to perform this test "During the startup test program."

The Monticello startup test program has been completed and is not required to be performed again. Thus, there is no need to retain this requirement in the ITS. This change is considered a presentation preference change only and, as such, is considered an administrative change.

MORE RESTRICTIVE CHANGES

M.1 CTS 4.3.E states that the reactivity anomaly Surveillance shall be performed "at each" startup following refueling outages. The ITS SR 3.1.2.1 Surveillance Frequency states that the Surveillance is performed "Once within 24 hours after reaching equilibrium conditions" following startup after fuel movement within the reactor pressure vessel or control rod replacement. This changes the CTS by providing an explicit time period to complete the Surveillance following a startup. This change to the "following refueling outage" portion of the frequency is discussed in DOC M.2.

The purpose of CTS 4.3.E is to verify the core reactivity after in-vessel operations, which could have significantly altered the core reactivity. A specific time for completing the reactivity anomaly surveillance CTS 4.3.E is proposed to clarify when "during the first startup" the test must be completed. This test is performed by comparing the difference between the actual control rod inventory and the predicted control rod inventory as a function of cycle exposure while at steady state reactor power conditions. Therefore, 24 hours after reaching these conditions is provided as a reasonable time to perform the required calculations and complete the appropriate verification, and thus this time is considered acceptable. Therefore, this change is considered a more restrictive change since a finite completion time is now provided.

M.2 CTS 4.3.E states, in part, that the reactivity anomaly Surveillance shall be performed "following refueling outages." This Frequency is changed in ITS SR 3.1.2.1 to be "after fuel movement within the reactor pressure vessel or control rod replacement." This changes the CTS by clearly defining the activities after which the reactivity anomaly Surveillance should be performed.

DISCUSSION OF CHANGES
ITS 3.1.2, REACTIVITY ANOMALIES

The purpose of CTS 4.3.E is to verify the core reactivity after in-vessel operations that could have altered the core reactivity. During refueling outages, core reactivity is normally significantly altered. However, conditions could arise mid-cycle that require replacing a fuel assembly or control rod, and the CTS would not require this Surveillance to be performed during the subsequent reactor startup. This mid-cycle replacement has the potential of altering core reactivity. The ITS words cover both planned refueling outages and other outages where CORE ALTERATIONS may occur, thus this change is considered acceptable. This change is considered a more restrictive change since the Surveillance will be required under more conditions than is currently required.

M.3 CTS 3.3.E requires the reactivity anomaly requirements to be met in the "reactor power operation" condition. ITS LCO 3.1.2 is Applicable in MODES 1 and 2. This changes the CTS by requiring the reactivity anomaly limit to be met in MODE 2 $\leq$ 1% RATED THERMAL POWER (RTP).

The purpose of CTS 3.3.E is to ensure plant operation is maintained within the assumptions of the safety analyses. This change expands the Applicability to require the reactivity anomaly limit to be met at all times when in MODE 2, instead of when > 1% RTP (the CTS 1.0.O definition states that Power Operation is when reactor power is > 1% RTP). This change is acceptable since the reactivity anomaly must be met in MODE 2 because control rods are typically being withdrawn during a startup. This change is designated as more restrictive because the LCO will be applicable under more reactor conditions.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.3.E states, in part, "At a specific steady state base condition" the reactor actual control rod inventory will be periodically compared to a "normalized computed" prediction of the inventory. CTS 3.3.E also implies that the reactivity difference shall be shall be within ± 1% $\Delta k/k$. CTS 4.3.E states, in part, the actual rod inventory shall be compared to a "normalized computed" prediction of inventory and that "These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle." Furthermore, the actual rod configuration will be compared to the configuration expected "based upon appropriately corrected past data." ITS LCO 3.1.2 states "The reactivity difference between the monitored control rod inventory and the predicted control rod inventory shall be within ± 1% $\Delta k/k$." ITS SR 3.1.2.1 states "Verify core reactivity difference between the monitored control rod inventory and the predicted control rod inventory is within ± 1% $\Delta k/k$." This changes the CTS by relocating these details for performing the reactivity anomaly Surveillance to the ITS Bases.

DISCUSSION OF CHANGES
ITS 3.1.2, REACTIVITY ANOMALIES

The removal of these details for evaluating surveillance requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS LCO 3.1.2 still retains the requirement that "The reactivity difference between the monitored control rod inventory and the predicted control rod inventory shall be within ± 1% Δk/k" and ITS SR 3.1.2.1 still retains the requirement to "Verify core reactivity difference between the monitored control rod inventory and the predicted control rod inventory is within ± 1% Δk/k." Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 - Relaxation of Required Action)* CTS 3.3.E states, in part, "If the difference exceeds one per cent, delta k, reactor power operation shall not be permitted until the cause has been evaluated and appropriate corrective action has been completed." This effectively requires an immediate unit shutdown if the reactivity difference is greater than 1% Δk/k. ITS 3.1.2 ACTIONS A and B cover the condition when the reactivity anomaly criterion is not met. ITS 3.1.2 ACTION A requires restoration of the core reactivity difference to within limit in 72 hours. If this Required Action and Completion Time are not met, ITS 3.1.2 ACTION B requires the unit to be in MODE 3 in 12 hours. This changes the CTS by allowing 72 hours to restore the reactivity difference before commencing a shutdown.

The purpose of the ITS 3.1.2 ACTIONS is to allow time to confirm that a reactivity anomaly is of no concern or to correct the problem. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period. The ITS 3.1.2 compensatory actions allow 72 hours of plant operations in MODE 1 and 2 before requiring a reactor shutdown. According to ITS 3.1.2 Required Action A.1 Bases restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This evaluation normally reviews the core conditions to determine their consistency with input to design calculations. Measured core and process parameters are also normally evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models may be reviewed to verify that they are adequate for representation of the core conditions. This change is acceptable

DISCUSSION OF CHANGES
ITS 3.1.2, REACTIVITY ANOMALIES

since the current requirement that does not allow reactor power operation to continue is overly restrictive because in most cases any reactivity anomaly is normally indicative of incorrect analysis inputs or assumptions of fuel reactivity used in the analysis. A determination and explanation of the cause of the anomaly would normally involve a fuel analysis department and the fuel vendor. Contacting and obtaining the necessary input may require a time period much longer than one shift (particularly on weekends and holidays). Since SHUTDOWN MARGIN has typically been demonstrated by test prior to reaching the conditions at which this Surveillance is performed, the safety impact of the extended time for evaluation is negligible. Given these considerations, the ITS allows this time to be 72 hours. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* The Frequency of the reactivity anomaly Surveillance in CTS 4.3.E is at least every "equivalent full power month" (approximately 611 MWD/T, where T is a short ton), and it is required to be performed "At specific power operating conditions." ITS SR 3.1.2.1 requires this same test to be performed every 1000 MWD/T during operations in MODE 1. This changes the CTS by extending the Surveillance Frequency from 611 MWD/T to 1000 MWD/T, and specifies that the "specific power operating condition" is MODE 1.

The purpose of CTS 4.3.E is to verify the reactivity difference is within limit. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. This changes extends the Surveillance Frequency for the reactivity anomaly test. This change is acceptable based on the slow rate of core reactivity changes due to fuel depletion and operating experience related to variations in core reactivity. The proposed change is consistent with the ISTS. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

control rod inventory

3.3.E LCO 3.1.2 The reactivity [difference] between the [monitored ~~rod density~~ and the predicted ~~rod density~~ shall be within ± 1% Δk/k. 1

APPLICABILITY: MODES 1 and 2.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.3.E	A. Core reactivity [difference] not within limit.	A.1 Restore core reactivity [difference] to within limit.	72 hours	1
3.3.E	B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours	

CTS

SURVEILLANCE REQUIREMENTS

CTS		SURVEILLANCE	FREQUENCY	
3.3.E, 4.3.E	SR 3.1.2.1	Verify core reactivity [ ] difference [ ] between the [ ] monitored ~~rod density~~ control rod inventory and the predicted ~~rod density~~ control rod inventory is within ± 1% Δk/k.	Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement AND 1000 MWD/T thereafter during operations in MODE 1	1

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.2, REACTIVITY ANOMALIES

1. The brackets have been removed and the proper plant specific information/value has been provided.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

INSERT 1

BASES

BACKGROUND

In accordance with GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity anomaly is used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity or control rod worth or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in assuring the reactor can be brought safely to cold, subcritical conditions. 1 2 ies

When the reactor core is critical ~~or in normal power operation~~, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers, producing zero net reactivity. 6

e.g., gadolinia 1

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel loaded in the previous cycles provide excess positive reactivity beyond that required to sustain steady state operation at the beginning of cycle (BOC). When the reactor is critical at RTP and operating moderator temperature, the excess positive reactivity is compensated by burnable absorbers ~~(if any)~~, control rods, and whatever neutron poisons (mainly xenon and samarium) are present in the fuel. The predicted core reactivity, as represented by control rod ~~density~~, is calculated by a 3D core simulator code as a function of cycle exposure. This calculation is performed for projected operating states and conditions throughout the cycle. The core reactivity is determined from control rod ~~densities~~ for actual plant conditions and is then compared to the predicted value for the cycle exposure. inventory 3

1

INSERT 1

In accordance with USAR, Section 1.2.2 (Ref. 1), the reactor core is designed so that control rod action, with the maximum worth control rod fully withdrawn and unavailable for use, is capable of bringing the reactor core subcritical and maintaining it so from any power level in the operating cycle. In addition, the reactor core and associated systems are designed to accommodate unit operational transients or maneuvers that might be expected without compromising safety and without fuel damage.

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BASES

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is ~~either~~ an ~~explicit or~~ implicit assumption in the accident analysis evaluations (Ref. 2). In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod drop accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity. 1

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted rod ~~density~~ for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict rod ~~density~~ may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured rod ~~density~~ from the predicted rod ~~density~~ that develop during fuel depletion may be an indication that the assumptions of the DBA and transient analyses are no longer valid, or that an unexpected change in core conditions has occurred.

3 control

Inventory

3 Inventory

ies

Reactivity ~~a~~nomalies satisf~~y~~ Criterion 2 of 10 CFR 50.36(c)(2)(ii). 2

LCO

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the "Nuclear Design Methodology" are larger than expected. A limit on the difference between the monitored and the predicted rod ~~density~~ of ± 1% Δk/k has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

3 control

3 Inventory

APPLICABILITY

In MODE 1, most of the control rods are withdrawn and steady state operation is typically achieved. Under these conditions, the comparison between predicted and monitored core reactivity provides an effective measure of the reactivity anomaly. In MODE 2, control rods are typically being withdrawn during a startup. In MODES 3 and 4, all control rods are fully inserted and therefore the reactor is in the least reactive state, where monitoring core reactivity is not necessary. In MODE 5, fuel loading

BASES

APPLICABILITY (continued)

results in a continually changing core reactivity. SDM requirements
(LCO 3.1.1) ensure that fuel movements are performed within the bounds 5
before or of the safety analysis, and an SDM demonstration is required during the
first startup following operations that could have altered core reactivity
(e.g., fuel movement, control rod replacement, shuffling). The SDM test,
required by LCO 3.1.1, provides a direct comparison of the predicted and
monitored core reactivity at cold conditions; therefore, reactivity anomaly 2
is not required during these conditions. ies

ACTIONS

A.1

Should an anomaly develop between measured and predicted core reactivity, the core reactivity difference must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This evaluation normally reviews the core conditions to determine their consistency with input to design calculations. Measured core and process parameters are also normally evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models may be reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

B.1

If the core reactivity cannot be restored to within the 1% Δk/k limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Verifying the reactivity difference between the monitored and predicted
3 control rod density is within the limits of the LCO provides added assurance that 3
plant operation is maintained within the assumptions of the DBA and
1 process computer transient analyses. The Core Monitoring System calculates the rod inventory
density for the reactor conditions obtained from plant instrumentation. A
comparison of the monitored rod density to the predicted rod density at
control 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the monitored and predicted control rod ~~density~~ inventory can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at ≥ 75% RTP have been obtained. INSERT 2 The 1000 MWD/T (where T is a short ton) Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1. 3 1

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 26~~ USAR, Section 1.2.2 1
2. U ~~F~~SAR, Chapter ~~[15]~~ 14. 1 4

1

INSERT 2

At a specific steady state base condition the actual control rod inventory will be periodically compared to a normalized computed prediction of the inventory. The comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle.

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.2 BASES, REACTIVITY ANOMALIES

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. Typographical/grammatical error corrected.

3. Changes are made to reflect those changes made to the Specification.

4. The brackets have been removed and the proper plant specific information/value has been provided.

5. Changes are made to be consistent with the Specification.

6. Editorial change made for clarity.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.2, REACTIVITY ANOMALIES

There are no specific NSHC discussions for this Specification.

ATTACHMENT 3

ITS 3.1.3, Control Rod OPERABILITY

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS | ITS

Add proposed LCO 3.1.3

Control Rod OPERABILITY

Add proposed ACTIONS Note

A.2 A.3 M.4

Add proposed SR 3.1.3.1 and SR 3.1.3.4

3.0 LIMITING CONDITIONS FOR OPERATION

2. ~~Reactivity margin - stuck control rods.~~

A.4

Add proposed ACTION A Note

ACTION A — M.1 — Add proposed Required Actions A.1

(a) Control rod drives which cannot be moved ~~with control rod drive pressure~~ shall be considered inoperable. The ~~directional control valves~~ for inoperable control rods shall be disarmed ~~electrically~~ and the rods shall be in such positions that Specification 3.3.A.1 is met.

2 hours

72 hours

LA.1

ACTION A — A.5 — Applicability: MODES 1 and 2

(b) If a partially or fully withdrawn control rod is stuck (i.e., cannot be moved with control rod drive or scram pressure), the reactor shall be brought to a hot shutdown condition within 48 hours. Shutdown is not required if it is confirmed that control rod drive collet housing failure is not the cause of the immovable control rod.

M.1 L.2 M.2

two or

ACTION B

(c) If more ~~than six~~ non-fully inserted control rods are inoperable during power operation, the reactor shall be brought to a hot shutdown condition within ~~48~~ hours. 12

M.3 A.2

A.2 — B. ~~Control Rod Withdrawal~~

A.4 — Add proposed Required Action C.1 Note

ACTION C — M.1 — 4 hours — LA.1 — Applicability: MODES 1 and 2 — L.6

1. ~~Each control rod shall be coupled to its drive~~ or completely inserted and the ~~directional control valves~~ disarmed ~~electrically~~. This requirement does not apply when removing a control rod drive for inspection as long as the reactor is in the refueling mode.

LA.1 — 3 hours — M.1

See ITS 3.10.5

Add proposed ACTION D — M.6

4.0 SURVEILLANCE REQUIREMENTS

~~2. Reactivity margin - stuck control rods.~~

L.1

Add proposed Note to SR 3.1.3.2 and SR 3.1.3.3 — L.3

SR 3.1.3.2, SR 3.1.3.3 — (a) Each fully or partially withdrawn operable control rod shall be ~~exercised~~ at least one notch each ~~week~~.

Inserted — M.5

31 days for SR 3.1.3.3 — L.3

Required Action A.3 — (b) If power operation is continuing with one fully or partially withdrawn control rod that is inoperable because it is stuck (i.e., cannot be moved by drive or scram pressure), each fully or partially withdrawn operable control rod shall be ~~exercised~~ at least one notch ~~every~~ 24 hour ~~period~~. This surveillance is not required if it has been confirmed that control rod drive collet housing failure is not the cause of the immovable control rod.

Inserted — M.5

from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint of the RWM

M.8 — L.4

~~(c) If power operation is continuing with two or more non-fully inserted control rods that are inoperable, each operable fully or partially withdrawn control rod shall be exercised at least one notch every 24 hour period.~~

M.2

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ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	B. Control Rod Withdrawal SR 3.1.3.5 1. The coupling integrity shall be verified for each withdrawn control rod as follows: each time (a) ~~when~~ the rod is fully withdrawn ~~the first time subsequent to each refueling outage~~, observe that the drive does not go to the overtravel position; and and prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling M.9

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A.1

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	(b) when the rod is withdrawn the first time subsequent to each refueling outage, observe discernible response of the nuclear instrumentation. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is critical shall be performed to observe nuclear instrumentation response. L.5
2. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all operable control rods are fully inserted and Specification 3.3.A.1 is met.	2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded. See CTS 3/4.3.B.2
3.(a) Control rod withdrawal sequences shall be established so that the maximum calculated reactivity that could be added by dropout of any increment of any one control blade will not make the core more than 1.3% Δk supercritical. See ITS 3.1.6	3.(a) To consider the rod worth minimizer operable, the following steps must be performed: (i) The control rod withdrawal sequence for the rod worth minimizer computer shall be verified as correct. (ii) The rod worth minimizer computer on-line diagnostic test shall be successfully completed. (iii) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified. See ITS 3.3.2.1

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A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
F. Scram Discharge Volume 1. During reactor operation, the scram discharge volume vent and drain valves shall be operable, except as specified below. 2. If any scam discharge volume vent or drain valve is made or found inoperable, the integrity of the scram discharge volume shall be maintained by either: a. Verifying daily, for a period not to exceed 7 days, the operability of the redundant valve(s), or b. Maintaining the inoperable valve(s), or the associated redundant valve(s), in the closed position. Periodically the inoperable and the redundant valve(s) may both be in the open position to allow draining the scram discharge volume. If a or b above cannot be met, at least all but one operable control rods (not including rods removed per specification 3.10.E or inoperable rods allowed by 3.3.A.2) shall be fully inserted within ten hours.	F. Scram Discharge Volume The scram discharge volume vent and drain valves sh be cycled quarterly. Once per operating cycle verify the scram discharge volume vent and drain valves close within 30 seconds after receipt of a reactor scram signal and open when the scram is reset.

See ITS 3.1.8

A.6

ACTION E

G. Required Action

1. If Specifications 3.3.A through D above are not met, an orderly shutdown shall be initiated and have reactor in the cold shutdown condition within 24 hours.

(except when the reactor mode switch is in the Refuel position)

A.6

12

MODE 3

L.6

Add proposed Condition E second part

M.7

DISCUSSION OF CHANGES
ITS 3.1.3, CONTROL ROD OPERABILITY

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3/4.3.A.2 provides requirements for stuck control rods. CTS 3/4.3.B.1 provides requirements for control rod coupling. ITS 3.1.3 provides requirements for each control rod. ITS LCO 3.1.3 states "Each control rod shall be OPERABLE." This changes the CTS by combining the OPERABILITY requirements for control rods into one Specification and adding an explicit statement concerning control rod OPERABILITY. Additional aspects of control rod OPERABILITY are also added in accordance with DOC M.4.

The purpose of ITS 3.1.3 is to include in one Specification all conditions that can affect the ability of the control rods to provide the necessary reactivity insertion. This change is acceptable because it provides a clear statement concerning the OPERABILITY requirements for each control rod. This change is acceptable since there are no technical changes. Therefore, this change is considered a presentation preference change only and, as such, is considered an administrative change.

A.3 CTS 3.3.A.2.(a) states that the directional control valves for inoperable control rods shall be disarmed. CTS 3.3.B.1 states that each control rod shall be coupled to its drive or completely inserted and the directional control valves disarmed. These CTS Actions do not limit the number of control rods to which these Actions apply. ITS 3.1.3 ACTIONS Note states "Separate Condition entry is allowed for each control rod." This changes the CTS by adding an explicit Note for separate condition entry for each control rod.

The purpose of CTS 3.3.A.2.(a) and CTS 3.3.B.1, in part, is to provide compensatory actions for an inoperable control rod on an individual basis. This change provides more explicit instructions for proper application of the ACTIONS for Technical Specification compliance. In conjunction with the proposed Specification 1.3, "Completion Times," this Note provides direction consistent with the intent of the existing ACTIONS for inoperable control rods. It is intended that each inoperable control rod be allowed a specified period of time in which compliance with certain limits is verified and, when necessary, the control rod is fully inserted and disarmed. Therefore, this change is considered a presentation preference change only and, as such, is considered an administrative change.

A.4 CTS 3.3.A.2.(a) states, in part, "The directional control valves for inoperable control rods shall be disarmed." CTS 3.3.B.1 states, in part, "Each control rod shall be coupled to its drive or completely inserted and the directional control valves disarmed." These compensatory actions are covered in ITS 3.1.3 ACTION A for stuck rods and ITS 3.1.3 ACTION C for coupling inoperabilities. In

DISCUSSION OF CHANGES
ITS 3.1.3, CONTROL ROD OPERABILITY

addition, these ITS 3.1.3 ACTIONS include a Note that states rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation. This changes the CTS by adding these clarification Notes.

The purpose of the ITS 3.1.3 ACTION A and ITS 3.1.3 Required Action C.1 Notes are to allow continued unit operation with inoperable control rods. This change is acceptable since CTS 3/4.3.B.3.(b) allows the RWM to be bypassed. To complete the associated actions the RWM may be required to be bypassed. This note is informative in that the RWM may be bypassed at any time, provided the proper ACTIONS of CTS 3/4.3.B.3.(b) (ITS 3.3.2.1), the RWM Specification, are taken. This is a human factors consideration to assure clarity of the requirement and allowance. Therefore, this change is considered a presentation preference change only and, as such, is considered an administrative change.

A.5 CTS 3.3.A.2 does not explicitly state when the stuck control rod requirements are required to be met. However, CTS 3.3.A.2.(b) states that the reactor should be brought to hot shutdown under certain situations. ITS 3.1.3 is applicable in MODES 1 and 2. This changes the CTS by explicitly stating the Applicability.

The purpose of CTS 3.3.A.2 is to limit the number of stuck control rods. This change is acceptable because the proposed MODE is consistent with the current shutdown condition. This change is considered a presentation preference change only and, as such, is considered an administrative change.

A.6 These changes to CTS 3.3.G are provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the USNRC for approval in NMC letter L-MT-05-013, from Thomas J. Palmisano (NMC) to USNRC, dated April 12, 2005. As such, these changes are administrative.

<u>MORE RESTRICTIVE CHANGES</u>

M.1 CTS 3.3.A.2.(a) states, in part, "The directional control valves for inoperable control rods shall be disarmed electrically and the rods shall be in such positions that Specification 3.3.A.1 is met." CTS 3.3.B.1 states, in part, "Each control rod coupled to its drive or completely inserted and the directional control valves disarmed." ITS 3.1.3 ACTION A covers the condition of one withdrawn control rod stuck, and requires the immediate verification that the stuck control rod separation criteria is met (Required Action A.1), the disarming of the associated control rod drive within 2 hours (Required Action A.2), and the performance of SR 3.1.1.1 (SHUTDOWN MARGIN verification test) within 72 hours (Required Action A.4). ITS 3.1.3 ACTION C covers the condition of one or more control rods inoperable for reasons other than a stuck control rod, and requires fully inserting an inoperable control rod within 3 hours (Required Action C.1) and disarming the associated control rod drive within 4 hours (Required Action C.2). This changes the CTS by adding finite times to perform the Required Actions and adds a new Required Action to verify stuck control rod separation criteria is met.

The purpose of CTS 3.3.A.2.(a) and CTS 3.3.B.1 are to place the unit in a safe condition when control rods are inoperable. This change is acceptable since the

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ITS 3.1.3, CONTROL ROD OPERABILITY

proposed Completion Times for performing the stuck control rod separation criteria verification, disarming control rod drives, inserting inoperable control rods, and for performing a SHUTDOWN MARGIN test are consistent with industry practice and can be safely accomplished. The stuck rod separation criteria is defined in the ITS 3.1.3 Bases. This additional requirement ensures the local scram reactivity will be met with a stuck rod. Disarming a control rod as required by CTS 3.3.A.2.(a) and CTS 3.3.B.1 involves personnel actions by other than control room operating personnel. This process will require coordination of personnel and preparation of equipment, and potentially require anti-contamination "dress-out," in addition to the actual procedure of disarming the control rod. The proposed Completion Times are acceptable in recognition of the potential time required to complete this task. The proposed time to disarm a control rod does not represent a significant safety concern as the control rod is already in an acceptable position and the ACTION to disarm is solely a mechanism for precluding the potential for damage to the control rod drive mechanism. With a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Also, a notch test is required by ITS 3.1.3 Required Action A.3 for each remaining withdrawn control rod to ensure that no additional control rods are stuck. Given these considerations, the time to demonstrate SHUTDOWN MARGIN in ITS 3.1.3 Required Action A.4 is 72 hours. This Completion Time provides a reasonable time to perform the analysis or test. The change has been designated as more restrictive because it adds an explicit Required Action Completion Times and adds a new Required Action to verify stuck rod separation criteria.

M.2 CTS 3.3.A.2.(c) allows continued operation with up to six non-fully inserted, inoperable (i.e., stuck) control rods. CTS 4.3.A.2.(c) states "If power operation is continuing with two or more non-fully inserted control rods that are inoperable, each operable fully or partially withdrawn control rod shall be exercised at least one notch every 24 hours." ITS 3.1.3 ACTION B requires the unit to be in MODE 3 with two stuck control rods. This changes the CTS by changing the number of non-fully inserted control rods that can be inoperable (i.e., stuck) and continue operations in MODE 1 and 2 from "six" to "one."

The purpose of CTS 3.3.A.2.(c) is to limit the number of non-fully inserted stuck control rods. This change is acceptable since with two or more withdrawn control rods stuck the unit must be brought to MODE 3. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The CTS 4.3.A.2.(c) requirement to test all OPERABLE fully or partially withdrawn control rods is not necessary since operation is not allowed with two or more non-fully inserted stuck control rods. This change is designated as more restrictive because it changes the number of non-fully inserted (i.e., stuck) inoperable control rods in MODE 1 and 2 from "six" to "one."

M.3 CTS 3.3.A.2.(c), in part, requires the unit to be in hot shutdown (MODE 3) in within 48 hours. ITS 3.1.3 ACTION B requires the unit to be in MODE 3 within 12 hours. This changes the CTS by changing the time to reach MODE 3 from 48 hours to 12 hours.

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The purpose of CTS 3.3.A.2.(c), in part, is to provide the appropriate time for the unit to be in MODE 3. This change is acceptable since the allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. This change is designated as more restrictive because it reduces the required time to achieve MODE 2 from 48 hours to 12 hours.

M.4 CTS 3/4.3.A.2 provides requirements for stuck control rods. CTS 3/4.3.B.1 provides requirements for control rod coupling. There are no requirements associated with the determination of each control rod position and maximum scram time of the control rods. ITS 3.1.3 includes two Surveillance Requirements to cover these requirements. ITS SR 3.1.3.1 requires the determination of the position of each control rod every 24 hours. ITS SR 3.1.3.4 requires the verification that each control rod scram time from the fully withdrawn position to notch position 06 is within limit (i.e. ≤ 7 seconds) in accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4. This changes the CTS by adding two additional OPERABILITY requirements for the control rods (i.e., maximum scram insertion time, and control rod position).

The purpose of the new Surveillance Requirements is to ensure important aspects of control rod OPERABILITY are monitored on a regular basis. This change is acceptable because it provides additional assurance that the control rods will provide its scram function (i.e., scram insertion time and its position will be known). This change is designated as more restrictive because it adds two new Surveillance Requirements to the CTS.

M.5 CTS 4.3.A.2.(a) requires each fully or partially withdrawn operable control rod to be "exercised" at least one notch. CTS 4.3.A.2.(b) requires the same testing when a control rod is found to be stuck. ITS SR 3.1.3.2, ITS SR 3.1.3.3, and ITS 3.1.3 Required Action A.3 requires the same testing however the control rods must be "inserted" in lieu of "exercised." This changes the CTS by requiring the OPERABLE withdrawn control rods to be "inserted" one notch instead of "exercised" one notch.

The purpose of CTS 4.3.A.2.(a) and CTS 4.3.A.2.(b) are to periodically verify that each withdrawn OPERABLE control rod is not stuck and is free to insert on a scram signal. This change is acceptable because it provides additional assurance that the control rods will provide their scram function. The existing requirement to exercise the control rod could be met by control rod withdrawal. It is conceivable that a mechanism causing binding of the control rod that prevents insertion could exist and that a withdrawal test would not detect the problem. Since the purpose of the test is to assure scram insertion capability, restricting the test to control rod insertion provides an increased likelihood of this test detecting a problem that impacts insertion capability. This change is designated as more restrictive because it changes the CTS acceptance criteria.

M.6 CTS 3.3.A.2 provides requirements for stuck control rods. CTS 3.3.B.1 provides requirements for control rod coupling. ITS 3.1.3 ACTION D provides an additional restriction for when two or more inoperable control rods are not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods and reactor power is

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ITS 3.1.3, CONTROL ROD OPERABILITY

≤ 10% RTP. In this condition, ITS 3.1.3 ACTION D requires within 4 hours either the restoring of compliance with BPWS or the restoring of a control rod to OPERABLE status. This changes the CTS by adding an explicit ACTION for inoperable control rods under certain conditions when reactor power is ≤ 10% RTP.

The purpose of ITS 3.1.3 ACTION D is to provide control rod operational restrictions at ≤ 10% RTP. This change is acceptable because out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a control rod drop accident. At ≤ 10% RTP, the generic BPWS analysis requires inserted control rods not in compliance with BPWS to be separated by at least two OPERABLE control rods in all directions, including the diagonal. Therefore, if two or more inoperable control rods are not in compliance with BPWS and not separated by at least two OPERABLE control rods, action must be taken to restore compliance with BPWS or restore the control rods to OPERABLE status. This change is designated as more restrictive because it adds a new ACTION to the CTS.

M.7 CTS 3/4.3.B.1 does not place a limitation of the number of inoperable control rods. ITS 3.1.3 ACTION E (second part of Condition E) covers the condition for nine or more inoperable control rods, and requires the unit to be in MODE 3 in 12 hours. This changes the CTS by adding an explicit ACTION for nine or more inoperable control rods.

The purpose of ITS 3.1.3 Condition E (second part) is to limit the number of inoperable control rods. The change is acceptable since this condition (with nine or more inoperable control rods) could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. This change is more restrictive since a limitation on the number of inoperable control rods has been added to the CTS.

M.8 CTS 4.3.A.2.(b), which requires a periodic exercise test of the remaining fully and partially withdrawn OPERABLE control rods when a control rod is found to be stuck, states "This surveillance is not required if it has been confirmed that control rod drive collet housing failure is not the cause of the immovable control rod." The ITS does not maintain this allowance. ITS 3.1.3 Required Action A.3 will require a similar test when a control rod is found to be stuck, regardless of the reason for the stuck control rod. This changes the CTS by requiring an insertion test of remaining fully and partially withdrawn OPERABLE control rods when a stuck rod is found, regardless of the reason the rod is stuck.

The purpose of CTS 4.3.A.2.(b) is to verify that each control rod is not stuck and is free to insert on a scram signal. This change is considered acceptable since the test will now be required regardless of the reason the rod is stuck. This will ensure that all remaining fully and partially withdrawn OPERABLE control rods are not also stuck. This change is a more restrictive change since the ITS will require a test under more conditions than currently required in the CTS.

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M.9 CTS 4.3.B.1.(a) states that "when the rod is fully withdrawn the first time subsequent to each refueling outage," observe that the drive does not go to the overtravel position. ITS SR 3.1.3.5 requires the same verification, however, it must be performed each time the control rod is withdrawn to the full out position and prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling. This changes the CTS by changing the requirement to perform the coupling verification from "when the rod is fully withdrawn the first time subsequent to each refueling outage" to "Each time the control rod is withdrawn to full out position" and by adding the new Frequency of "Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling."

The purpose of CTS 4.3.B.1.(a) is to ensure each control rod is coupled to its associated drive. The requirement to perform the coupling verification "when the rod is fully withdrawn the first time subsequent to each refueling outage" has been changed to "Each time the control rod is withdrawn to "full out" position" and a new Frequency, "Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling," has been added. This change is acceptable because a coupling check is necessary after any work is performed on a control rod or Control Rod Drive System that could affect coupling. In addition, the requirement to perform the Surveillance each time the control rod is withdrawn is acceptable since a control rod could uncouple from its drive whenever a control rod is moved, not just after the first time it is fully withdrawn subsequent to each refueling outage. If a control rod is inserted one notch or more and then returned to the "full out" position during the performance of ITS SR 3.1.3.2 or for some other reason, a coupling verification can be easily performed since the verification only requires a check to make sure the control rod does no go to the withdrawn overtravel position. This change is designated as more restrictive because it requires the control rod coupling test to be verified more often.

<u>RELOCATED SPECIFICATIONS</u>

None

<u>REMOVED DETAIL CHANGES</u>

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.3.A.2.(a) states, in part, the "directional control valves" for inoperable control rods shall be disarmed "electrically." CTS 3.3.B.1 states, in part, each control rod shall be coupled to its drive or completely inserted and the "directional control valves" disarmed "electrically." ITS 3.1.3 ACTION A covers the condition of one withdrawn control rod stuck. ITS 3.1.3 Required Action A.2 states "Disarm the associated control rod drive (CRD)." ITS 3.1.3 ACTION C covers the condition of one or more control rods inoperable for reasons other than a stuck rod. ITS 3.1.3 Required Action C.2 states "Disarm the associated CRD." Neither of these two Required Actions provides specific details of how to disarm the CRD. This changes the CTS by relocating the

details that the "directional control valves" are disarmed "electrically" to the ITS Bases.

The removal of these details for performing Required Actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS 3.1.3 Required Actions A.2 and C.2 still retain the requirement to disarm the CRD. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.3.A.2.(a) states, in part, that control rod drives which cannot be moved "with control rod drive pressure" shall be considered inoperable. ITS 3.1.3 does not include this specific requirement. ITS 3.1.3 requires each control rod to be OPERABLE. A rod is considered OPERABLE, with respect to motion, if it can be inserted at least one notch using either scram pressure or normal control rod drive pressure (ITS SR 3.1.3.2 and SR 3.1.3.3) and, if it can be scrammed within ≤ 7.0 seconds (ITS SR 3.1.3.4). This changes the CTS by deleting the requirement to consider a control rod inoperable if it cannot be moved by control rod drive pressure alone.

The purpose of the ITS 3.1.3 is to base the OPERABILITY of an individual control rod on a combination of factors, including the scram insertion times. As long as a control rod can be scrammed within ≤ 7 seconds it should be considered to be OPERABLE. The control rod can satisfy this requirement with either accumulator pressure (scram pressure), control rod drive pressure, or a combination of the two. Accumulator OPERABILITY is addressed by LCO 3.1.5, "Control Rod Accumulators." The associated scram accumulator status for a control rod only affects the scram insertion times; therefore, an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the design basis accident and transient analyses. This reactivity control requirement is monitored in LCO 3.1.4, "Control Rod Scram Times." This change is acceptable since the proposed requirements will continue to ensure the reactivity control is maintained. This change is designated as less restrictive because a control rod will not be considered inoperable if it cannot be moved by control rod drive pressure alone.

L.2 *(Category 4 – Relaxation of Required Action)* CTS 3.3.A.2.(b) requires, in part, the unit to be in hot shutdown within 48 hours if it is confirmed that a control rod drive collet housing failure is the cause of the stuck control rod. ITS 3.1.3

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ITS 3.1.3, CONTROL ROD OPERABILITY

ACTION A covers the condition for one stuck control rod. Continuous operation is allowed regardless of the reason for the control rod being stuck. This changes the CTS by allowing continuous operation with any type of stuck rod even as a result of a control rod drive collet housing failure.

The purpose of CTS 3.3.A.2.(b) is to allow continuous unit operation with stuck rods as long as the cause of the failure is not a control rod drive collet housing failure. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a design basis accident occurring during the repair period. This change will allow continuous operation with one stuck rod regardless of the cause of the failure. ITS 3.1.3 ACTION A will allow continuous operation as long as it is verified that the stuck control rod separation criteria are met, the stuck rod is disarmed, the other control rods are confirmed to not be stuck, and SHUTDOWN MARGIN is met with the stuck rod. The control rod separation criteria is described in the ITS 3.1.3 Bases. The separation criteria are not met if: a) the stuck control rod occupies a location adjacent (face or diagonal) to two "slow" control rods, b) the stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) if the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods adjacent to one another. A "slow" control rod is described in the ITS 3.1.4 Bases. These proposed Required Actions are acceptable because they support continuous operation regardless of the type of failure since they continue to ensure SHUTDOWN MARGIN (SDM) can be met, other control rods are not stuck, and there is sufficient reactivity insertion capability to support the accident and transient analysis. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.3 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.3.A.2.(a) requires each fully or partially withdrawn operable control rod to be exercised at least one notch "each week." ITS SR 3.1.3.2 requires a similar Surveillance for fully withdrawn control rods and ITS SR 3.1.3.3 requires a similar Surveillance for partially withdrawn control rods, however the Surveillance Frequency for ITS SR 3.1.3.3 is every 31 days. In addition, each Surveillance contains a Note that allows the performance of the Surveillance to be delayed for a certain time after the control rod is withdrawn and THERMAL POWER is greater than the low power setpoint (LPSP) of the rod worth minimizer (RWM). ITS SR 3.1.3.2 may be delayed for 7 days while ITS SR 3.1.3.3 may be delayed 31 days. This changes the CTS by extending the Surveillance Frequency from 7 days to 31 days for control rods that are partially withdrawn and provides a delay period for initial performance of the Surveillance after a control rod is withdrawn and THERMAL POWER is greater than the LPSP of RWM.

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ITS 3.1.3, CONTROL ROD OPERABILITY

The purpose of CTS 4.3.A.2.(a) is to periodically verify that each withdrawn control rod is not stuck and is free to insert on a scram signal. This change is acceptable because the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Decreasing the Frequency of control rod exercise test for partially withdrawn control rods is acceptable based on the potential power reduction required to allow the control rod movement and considering the operating experience related to changes in control rod drive performance. These Surveillances are not required when THERMAL POWER is less than or equal to the actual LPSP of the RWM, since the notch insertions may not be compatible with the requirements of the Banked Position Withdrawal Sequence (BPWS) (LCO 3.1.6) and the RWM (LCO 3.3.2.1). This portion of the change is acceptable since the unit does not normally operate for extended periods below the LPSP and since during a startup the control rods are withdrawn which helps to verify the control rod is not stuck since control rods are being withdrawn. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.4 *(Category 3 – Relaxation of Completion Time)* CTS 4.3.A.2.(b) states, in part, "each fully or partially withdrawn operable control rod shall be exercised at least one notch every 24 hour period" when a control rod is found to be stuck. When a control rod is stuck, ITS 3.1.3 Required Action A.3 states to perform SR 3.1.3.2 and SR 3.1.3.3 (the control rod insertion Surveillances for fully and partially withdrawn control rods) for each withdrawn OPERABLE control rod "24 hours from discovery of the stuck rod concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM." This changes the CTS by only requiring the test to be performed one time, and allows the test to be delayed up to 24 hours from discovery of the stuck rod concurrent with THERMAL POWER greater than the LPSP of the RWM.

The purpose of CTS 4.3.A.2.(b) is to periodically verify that each withdrawn control rod is not stuck and is free to insert on a scram signal. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a design basis accident occurring during the repair period. This change, in part, only requires the test to be performed one time at the accelerated Frequency. This is acceptable since performing this test one time ensures the control rods are not stuck and are free to insert on a scram signal. ITS SR 3.1.3.2 and ITS SR 3.1.3.3, the control rod insertion Surveillances, are still required to be performed at the specified Surveillance Frequency when the unit is operating with a stuck rod. These tests will ensure the control rods are OPERABLE. This change also allows the tests to be delayed up to 24 hours from discovery of the stuck rod concurrent with THERMAL POWER greater than the LPSP of the RWM, since the notch insertions may not be compatible with the requirements of the Banked Position Withdrawal Sequence (BPWS) (LCO 3.1.6) and the RWM (LCO 3.3.2.1). This portion of the change is acceptable since the

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ITS 3.1.3, CONTROL ROD OPERABILITY

unit does not normally operate for extended periods below the LPSP and since during a startup the control rods are withdrawn which helps to verify the control rod is not stuck. This change is designated as less restrictive because the test only has to be performed once and the test may be delayed up to 24 hours from discovery of the stuck rod concurrent with THERMAL POWER greater than the LPSP of the RWM.

L.5 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.3.B.1.(b) states "when the rod is withdrawn the first time subsequent to each refueling outage, observe discernible response of the nuclear instrumentation. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is critical shall be performed to observe nuclear instrumentation response." ITS 3.1.3 does not include this requirement. This changes the CTS by eliminating the Surveillance Requirement to verify discernible nuclear instrumentation response when the rod is withdrawn.

The purpose of CTS 4.3.B.1.(b) is to ensure the control rod is coupled to its drive during the withdrawal of a control rod. This change is acceptable because the deleted Surveillance Requirement is not necessary to ensure the control rods are coupled to their drives. Coupling verification is performed to ensure each control rod is connected to its drive so that it will perform its intended function when necessary. ITS SR 3.1.3.5 requires verifying a control rod does not go to the withdrawn overtravel position. The overtravel position feature provides a positive check on the coupling integrity since only an uncoupled control rod drive can reach the overtravel position. The verification is required to be performed any time a control rod is withdrawn to the "full out" position (notch position 48) or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved and operating experience related to uncoupling events. Observation of nuclear instrumentation indication will still normally occur during control rod withdrawal since nuclear instrumentation indication is close to the controls used to withdraw control rods. This change is designated as less restrictive because a Surveillance that is required in the CTS will not be required in the ITS.

L.6 *(Category 2 – Relaxation of Applicability)* CTS 3.3.B.1 does not explicitly state when the control rod coupling requirements are required to be met, however it does state that the requirement is not applicable when moving a control rod drive for inspection as long as the reactor is in the refueling mode. CTS 3.3.G.1 requires the unit to be in cold shutdown (MODE 4) within 24 hours when the requirements of CTS 3/4.3.B.1 are not met. Thus, the implication is that CTS 3.3.B.1 is applicable in MODES 1, 2, and 3. ITS 3.1.3 states that the control rods must be OPERABLE in MODES 1 and 2 and ITS 3.1.3 ACTION E only requires the unit to be in MODE 3 (hot shutdown) within 12 hours when the actions are not met. This changes the CTS by only requiring the control rod coupling requirements to be met in MODES 1 and 2 and, concurrently, changes the shutdown action condition from cold shutdown (MODE 4) in 24 hours to hot shutdown (MODE 3) in 12 hours.

The purpose of CTS 3.3.B.1 is to ensure each control rod is coupled to its drive prior to control rod withdrawal to help ensure a control rod drop accident does not

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occur during plant operation. This change is acceptable because the requirements continue to ensure that the control rods are maintained in the MODES and other specified conditions assumed in the safety analyses. The control rods are not required to be OPERABLE in MODES 3 and 4 since the Reactor Manual Control System places a rod withdrawal block when the mode switch is placed in shutdown so that no control rods can be withdrawn. In this condition, all control rods will be inserted, therefore coupling requirements are not necessary since the potential for a rod drop accident is highly unlikely. During refueling (MODE 5), with the MODE switch in the refueling position, coupling requirements are not necessary since only one rod can be withdrawn at a time. The CTS 3.3.G.1 requirement to be in cold shutdown has been replaced with the requirement to be in MODE 3. Since the OPERABILITY requirements are only necessary in MODES 1 and 2, the necessary shutdown action condition is MODE 3. Consistent with other actions to be in MODE 3; 12 hours is provided to reach this MODE. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

3.3.A.2, 3.3.B.1 LCO 3.1.3 Each control rod shall be OPERABLE.

DOCs A.6, L.6 APPLICABILITY: MODES 1 and 2.

ACTIONS

---NOTE---
DOC A.3 Separate Condition entry is allowed for each control rod.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck. (3.3.A.2.(a), 3.3.A.2.(b))	---NOTE--- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation.	
	A.1 Verify stuck control rod separation criteria are met.	Immediately
	AND	
	A.2 Disarm the associated control rod drive (CRD).	2 hours
	AND	

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
4.3.A.2.(b)		A.3 Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
		AND	
3.3.A.2.(a)		A.4 Perform SR 3.1.1.1.	72 hours
3.3.A.2.(c)	B. Two or more withdrawn control rods stuck.	B.1 Be in MODE 3.	12 hours
3.3.B.1	C. One or more control rods inoperable for reasons other than Condition A or B.	C.1 -------------NOTE------------ RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. ------------------------------- Fully insert inoperable control rod.	3 hours
		AND	
		C.2 Disarm the associated CRD.	4 hours

CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME	
DOC M.6	D. ------------NOTE------------ Not applicable when THERMAL POWER > [10]% RTP. ------------------------------- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	D.1	Restore compliance with BPWS.	4 hours	1
		OR			
		D.2	Restore control rod to OPERABLE status.	4 hours	
	~~E. ------------NOTE------------ [Not applicable when THERMAL POWER > [10]% RTP. ------------------------------- One or more groups with four or more inoperable control rods.~~	~~E.1~~	~~Restore control rod to OPERABLE status.~~	~~4 hours]~~	2
3.3.G.1	F E Required Action and associated Completion Time of Condition A, C, D, or E not met. OR	F E.1	Be in MODE 3.	12 hours	2
DOC M.7	Nine or more control rods inoperable.				

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
DOC M.4	SR 3.1.3.1	Determine the position of each control rod.	24 hours	
4.3.A.2.(a)	SR 3.1.3.2	-------------------------NOTE------------------------- Not required to be performed until 7 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of RWM. -- Insert each fully withdrawn control rod at least one notch.	7 days	
4.3.A.2.(a)	SR 3.1.3.3	-------------------------NOTE------------------------- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. -- Insert each partially withdrawn control rod at least one notch.	31 days	
DOC M.4	SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to notch position [06] is ≤ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4	1
4.3.B.1	SR 3.1.3.5	Verify each control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position **AND** Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling	

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.3, CONTROL ROD OPERABILITY

1. The brackets have been removed and the proper plant specific information has been provided.

2. As stated in the ISTS Bases, ISTS 3.1.3 ACTION E is applicable to plants with ANF fuel. Monticello does not have this type of fuel. Consequently, this ACTION is not applicable to Monticello and has been deleted. As a result of this deletion, the following ACTION has been renumbered.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Control Rod OPERABILITY

BASES

BACKGROUND

Control rods are components of the Control Rod Drive (CRD) System, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure under conditions of normal operation, including anticipated operational occurrences, that specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements of GDC 26, ~~GDC 27,~~ GDC 28, and 29 (Ref. 1). as described in USAR, Section 1.2.2 1

121

The CRD System consists of ~~137~~ locking piston control rod drive mechanisms (CRDMs) and a hydraulic control unit for each drive mechanism. The locking piston type CRDM is a double acting hydraulic piston, which uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion. 1

and LCO 3.1.6, "Rod Pattern Control,"

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," ~~and~~ 1 LCO 3.1.5, "Control Rod Scram Accumulators," ensure that the 1 performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in the evaluations involving control rods are presented in References 2, 3, and 4. The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical and for limiting the potential effects of reactivity insertion events caused by malfunctions in the CRD System.

The capability to insert the control rods provides assurance that the assumptions for scram reactivity in the DBA and transient analyses are not violated. Since the SDM ensures the reactor will be subcritical with the highest worth control rod withdrawn (assumed single failure), the

BASES

APPLICABLE SAFETY ANALYSES (continued)

additional failure of a second control rod to insert, if required, could invalidate the demonstrated SDM and potentially limit the ability of the CRD System to hold the reactor subcritical. If the control rod is stuck at an inserted position and becomes decoupled from the CRD, a control rod drop accident (CRDA) can possibly occur. Therefore, the requirement that all control rods be OPERABLE ensures the CRD System can perform its intended function.

The control rods also protect the fuel from damage which could result in release of radioactivity. The limits protected are the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), and the fuel ~~damage~~ design limit (see Bases for LCO 3.1.6~~, "Rod Pattern Control"~~) during reactivity insertion events. 1 2

The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against violating fuel ~~damage~~ design limits during a CRDA. The Bases for LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6 discuss in more detail how the SLs are protected by the CRD System. 1

Control rod OPERABILITY satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The OPERABILITY of an individual control rod is based on a combination of factors, primarily, the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram accumulator status for a control rod only affects the scram insertion times; therefore, an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

INSERT 1 1

1

INSERT 1

OPERABILITY requirements for control rods also include correct assembly of the CRD housing supports.

BASES

APPLICABILITY	In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."
ACTIONS	The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each control rod. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable control rod. Complying with the Required Actions may allow for continued operation, and subsequent inoperable control rods are governed by subsequent Condition entry and application of associated Required Actions.

A.1, A.2, A.3, and A.4

A control rod is considered stuck if it will not insert by either CRD drive water or scram pressure. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Actions are modified by a Note, which allows the rod worth minimizer (RWM) to be bypassed if required to allow continued operation. LCO 3.3.2.1, "Control Rod Block Instrumentation," provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, the local scram reactivity rate assumptions may not be met if the stuck control rod separation criteria are not met. Therefore, a verification that the separation criteria are met must be performed immediately. The separation criteria are not met if: a) the stuck control rod occupies a location adjacent to two "slow" control rods, b) the stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) if the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods adjacent to one another. The description of "slow" control rods is provided in LCO 3.1.4, "Control Rod Scram Times." In addition, the associated control rod drive must be disarmed in 2 hours. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable time to perform the Required Action in an orderly manner. Isolating the control rod from scram prevents damage to the CRDM. The control rod can be isolated from scram and ~~normal insert~~ and withdraw pressure, yet still maintain cooling water to the CRD.

stuck control rod 2

(face or diagonal) 2

elsewhere in the core 2

INSERT 2

2

is maintained

and normal insert and withdraw pressure

isolation method should also ensure

2

INSERT 2

The control rod must be isolated from both scram and normal insert and withdraw pressure.

BASES

ACTIONS (continued)

Monitoring of the insertion capability of each withdrawn control rod must also be performed within 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM. SR 3.1.3.2 and SR 3.1.3.3 perform periodic tests of the control rod insertion capability of withdrawn control rods. Testing each withdrawn control rod ensures that a generic problem does not exist. This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." The Required Action 3 A.~~2~~ Completion Time only begins upon discovery of Condition A concurrent with THERMAL POWER greater than the actual LPSP of the RWM since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The allowed Completion Time of 24 hours from discovery of Condition A, concurrent with THERMAL POWER greater than the LPSP of the RWM, provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests. 3

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to preserve the single failure criterion, an additional control rod would have to be assumed to fail to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

The allowed Completion Time of 72 hours to verify SDM is adequate, considering that with a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram. Even with the postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach ~~and maintain~~ MODE 3 conditions ~~(Ref. 5)~~. 1

BASES

ACTIONS (continued)

B.1

With two or more withdrawn control rods stuck, the plant must be brought to MODE 3 within 12 hours. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

a With one or more control rods inoperable for reasons other than being stuck in ~~the~~ withdrawn position, operation may continue, provided the control rods are fully inserted within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. The control rods can be electrically disarmed by disconnecting power from all four directional control valve solenoids. Required Action C.1 is modified by a Note, which allows the RWM to be bypassed if required to allow insertion of the inoperable control rods and continued operation. LCO 3.3.2.1 provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis. 2

The allowed Completion Times are reasonable, considering the small number of allowed inoperable control rods, and provide time to insert and disarm the control rods in an orderly manner and without challenging plant systems.

D.1 and D.2

Out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. At ≤ 10% RTP, the generic banked position withdrawal sequence (BPWS) analysis (Ref. 5) requires inserted control rods not in compliance with BPWS to be separated by at least two OPERABLE control rods in all directions, including the diagonal (i.e., all other control rods in a five-by-five array centered on the inoperable control rod are OPERABLE). 2
Therefore, if two or more inoperable control rods are not in compliance

BASES

ACTIONS (continued)

with BPWS and not separated by at least two OPERABLE control rods, in all directions, action must be taken to restore compliance with BPWS or restore the control rods to OPERABLE status. Condition D is modified by a Note indicating that the Condition is not applicable when > 10% RTP, since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.6. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring. 2

~~E.1~~

~~In addition to the separation requirements for inoperable control rods, an assumption in the CRDA analysis for ANF fuel is that no more than three inoperable control rods are allowed in any one BPWS group. Therefore, with one or more BPWS groups having four or more inoperable control rods, the control rods must be restored to OPERABLE status. Required Action E.1 is modified by a Note indicating that the Condition is not applicable when THERMAL POWER is > 10% RTP since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.6. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring.~~ 4

E ~~F~~.1 4

If any Required Action and associated Completion Time of Condition A, C, D, ~~or E~~ are not met, or there are nine or more inoperable control rods, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. This ensures all insertable control rods are inserted and places the reactor in a condition that does not require the active function (i.e., scram) of the control rods. The number of control rods permitted to be inoperable when operating above 10% RTP (e.g., no CRDA considerations) could be more than the value specified, but the occurrence of a large number of inoperable control rods could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. 4

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The position of each control rod must be determined to ensure adequate information on control rod position is available to the operator for determining ~~CRD~~ control rod OPERABILITY and controlling rod patterns. Control rod position may be determined by the use of OPERABLE position indicators, by moving control rods to a position with an OPERABLE indicator, or by the use of other appropriate methods. The 24 hour Frequency of this SR is based on operating experience related to expected changes in control rod position and the availability of control rod position indications in the control room. 5

SR 3.1.3.2 and SR 3.1.3.3

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal. These Surveillances are not required when THERMAL POWER is less than or equal to the actual LPSP of the RWM, since the notch insertions may not be compatible with the requirements of the Banked Position Withdrawal Sequence (BPWS) (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The 7 day Frequency of SR 3.1.3.2 is based on operating experience related to the changes in CRD performance and the ease of performing notch testing for fully withdrawn control rods. Partially withdrawn control rods are tested at a 31 day Frequency, based on the potential power reduction required to allow the control rod movement and considering the large testing sample of SR 3.1.3.2. Furthermore, the 31 day Frequency takes into account operating experience related to changes in CRD performance. At any time, if a control rod is immovable, a determination of that control rod's trippability (OPERABILITY) must be made and appropriate action taken.

INSERT 3 3

SR 3.1.3.4

Verifying that the scram time for each control rod to notch position 06 is ≤ 7 seconds provides reasonable assurance that the control rod will insert when required during a DBA or transient, thereby completing its shutdown function. This SR is performed in conjunction with the control rod scram time testing of SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1, "Reactor

3

INSERT 3

These SRs are modified by Notes that allow 7 days and 31 days respectively, after withdrawal of the control rod and increasing power to above the LPSP, to perform the Surveillance. This acknowledges that the control rod must be first withdrawn and THERMAL POWER must be increased to above the LPSP before performance of the Surveillance, and therefore, the Notes avoid potential conflicts with SR 3.0.1 and SR 3.0.4.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Protection System (RPS) Instrumentation," and the functional testing of SDV vent and drain valves in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlap this Surveillance to provide complete testing of the assumed safety function. The associated Frequencies are acceptable, considering the more frequent testing performed to demonstrate other aspects of control rod OPERABILITY and operating experience, which shows scram times do not significantly change over an operating cycle.

SR 3.1.3.5

Coupling verification is performed to ensure the control rod is connected to the CRDM and will perform its intended function when necessary. The Surveillance requires verifying that a control rod does not go to the withdrawn overtravel position when it is fully withdrawn. The overtravel position feature provides a positive check on the coupling integrity since only an uncoupled CRD can reach the overtravel position. The verification is required to be performed any time a control rod is withdrawn to the "full out" position (notch position 48) or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This includes control rods inserted one notch and then returned to the "full out" position during the performance of SR 3.1.3.2. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved and operating experience related to uncoupling events. (2) (3)

REFERENCES

1. 10 ~~CFR 50, Appendix~~ A, ~~GDC 26,~~ GDC 27, ~~GDC 28, and GDC 29~~. USAR, Section 1.2.2 (1)

(1) U

2. [F]SAR, ~~Section~~ [4.2.3.2.2.4]. Chapter 14 (6)

3. [F]SAR, ~~Section~~ [5A.4.3]. Chapter 14A (6)

4. [F]SAR, ~~Section~~ [15.1]. Chapter 3 (6)

5. NEDO-21231, "Banked Position Withdrawal Sequence," Section 7.2, January 1977.

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.3 BASES, CONTROL ROD OPERABILITY

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. Change made to be consistent with the Specification.

4. Changes are made to reflect changes made to the Specification.

5. Typographical/grammatical error corrected.

6. The brackets have been removed and the proper plant specific information has been provided.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.3, CONTROL ROD OPERABILITY

There are no specific NSHC discussions for this Specification.

ATTACHMENT 4

ITS 3.1.4, Control Rod Scram Times

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS | ITS

M.1

3.0 LIMITING CONDITIONS FOR OPERATION

C. Scram Insertion Times

LCO 3.1.4 and Table 3.1.4-1

Table 3.1.4-1 footnote (a)

Applicability

M.4

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

Percent of Rod Length Inserted	Seconds
5	0.398
20	0.954
50	2.120
90	3.80

Add proposed LCO 3.1.4 and Table 3.1.4-1 including Notes 1 and 2 and footnote b

M.1

4.0 SURVEILLANCE REQUIREMENTS

C. Scram Insertion Times

Add proposed Surveillance Requirement Note

M.2

SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, SR 3.1.4.4

During each Operating Cycle, each operable control rod shall be subjected to scram time tests from the fully withdrawn position. If testing is not accomplished during reactor power operation, the measured scram insertion times shall be extrapolated to the reactor power operation condition utilizing previously determined correlations.

Add proposed SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

M.2

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A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
F. Scram Discharge Volume 1. During reactor operation, the scram discharge volume vent and drain valves shall be operable, except as specified below. 2. If any scam discharge volume vent or drain valve is made or found inoperable, the integrity of the scram discharge volume shall be maintained by either: a. Verifying daily, for a period not to exceed 7 days, the operability of the redundant valve(s), or b. Maintaining the inoperable valve(s), or the associated redundant valve(s), in the closed position. Periodically the inoperable and the redundant valve(s) may both be in the open position to allow draining the scram discharge volume. If a or b above cannot be met, at least all but one operable control rods (not including rods removed per specification 3.10.E or inoperable rods allowed by 3.3.A.2) shall be fully inserted within ten hours.	F. Scram Discharge Volume The scram discharge volume vent and drain valves shall be cycled quarterly. Once per operating cycle verify the scram discharge volume vent and drain valves close within 30 seconds after receipt of a reactor scram signal and open when the scram is reset.

See ITS 3.1.8

A.3 1.

G. Required Action

ACTION A

If Specifications 3.3.A through D above are not met, an orderly shutdown shall be initiated and have reactor in the cold shutdown condition within 24 hours.

(except when the reactor mode switch is in the Refuel position) A.3

12 M.3

MODE 3 A.2

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DISCUSSION OF CHANGES
ITS 3.1.4, CONTROL ROD SCRAM TIMES

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 When the scram time requirements of CTS 3.3.C are not met, CTS 3.3.G.1 requires the unit to be in cold shutdown (MODE 4) within 24 hours. ITS 3.1.4 ACTION A only requires a shutdown to MODE 3. This changes the CTS by stating the unit must be shut down to MODE 3 instead of to MODE 4. The change to the time allowed to reach the required shutdown condition is discussed in DOC M.3.

The purpose of CTS 3.3.G.1 is to place the unit in a condition where the scram time limits of the control rods are not required. This change is considered acceptable since CTS 3.3.C is only Applicable in the power operation condition, i.e., MODES 1 and 2. Thus, once MODE 3 is achieved, continuation to MODE 4 is no longer required. Therefore, this change is considered administrative since the technical requirements are not being changed; the change is being made to ensure the shutdown actions are consistent with the current Applicability.

A.3 These changes to CTS 3.3.G are provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the USNRC for approval in NMC letter L-MT-05-013, from Thomas J. Palmisano (NMC) to USNRC, dated April 12, 2005. As such, these changes are administrative.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.3.C.1 specifies criteria for the average scram insertion time of all OPERABLE control rods from the fully withdrawn position to the 5%, 20%, 50%, and 90% inserted positions. CTS 3.3.C.2 specifies criteria for the average scram insertion time for the three fastest control rods of all groups of four control rods in a two by two array from the fully withdrawn position to the 5%, 20%, 50%, and 90% inserted positions. ITS LCO 3.1.4 states "a. No more than 8 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1, and " b. No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations." ITS Table 3.1.4-1 specifies the maximum scram times for each control rod, when reactor steam dome pressure is $\geq$ 800 psig, to notch positions 46, 36, 26, and 06. ITS Table 3.1.4-1 Note 1 states that OPERABLE control rods with scram times not within the limits of this Table are considered "slow." ITS Table 3.1.4-1 Note 2 states "Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with ITS SR 3.1.3.4, and are not considered "slow." ITS Table 3.1.4-1 footnote (b) states "Scram times as a function of reactor steam dome pressure when < 800 psig are

DISCUSSION OF CHANGES
ITS 3.1.4, CONTROL ROD SCRAM TIMES

within established limits." This changes the CTS by specifying control rod scram time for each individual control rod as a function of reactor steam dome pressure instead of the current scram time requirements based on the average scram insertion time of all OPERABLE control rods and for the average scram insertion time for the three fastest control rods of all groups of four control rods in a two by two array. In addition, criteria has been established for no more than 8 "slow" OPERABLE control rods and no more than 2 "slow" OPERABLE control rods occupying adjacent locations.

The purpose of the control rod scram time LCOs (CTS 3.3.C.1 and 3.3.C.2) is to ensure the negative scram reactivity is consistent with those values assumed in the accident and transient analysis. CTS 3.3.C.1 and 3.3.C.2 place requirements on the average scram times and local scram times (four control rod group). Because of the methodology used in the design basis transient analysis (one-dimensional neutronics), all control rods are assumed to scram at the same speed, which is the analytical scram time requirement. Performing an evaluation assuming all control rods scram at the analytical limit results in the generation of a scram reactivity versus time curve, the analytical scram reactivity curve. The purpose of the scram time LCO is to ensure that, under allowed unit conditions, this analytical scram reactivity will be met. Since scram reactivity cannot be readily measured at the unit, the safety analyses use appropriately conservative scram reactivity versus insertion fraction curves to account for the variation in scram reactivity during a cycle. Therefore, the Technical Specifications must only ensure the scram times are satisfied. The first obvious result is that, if all control rods scram at least as fast as the analytical limit, the analytical scram reactivity curve will be met. However, a distribution of scram times (some slower and some faster than the analytical limit) can also provide adequate scram reactivity. By definition, for a situation where all control rods do not satisfy the analytical scram time limits, the condition is acceptable if the resulting scram reactivity meets or exceeds the analytical scram reactivity curve. This can be evaluated using models that allow for a distribution of scram speeds. It follows that the more control rods that scram slower than the analytical limit, the faster the remaining control rods must scram to compensate for the reduced scram reactivity rate of the slower control rods. ITS 3.1.4 incorporates this philosophy by specifying scram time limits for each individual control rod instead of limits on the average of all control rods and the average of three fastest rods in all four control rod groups. This philosophy has been endorsed by the BWR Owners' Group and is described in EAS-46-0487, "Revised Reactivity Control Systems Technical Specifications." The scram time limits listed in ITS Table 3.1.4-1 have margin to the analytical scram time limits listed in EAS-46-0487, Table 3-4 to allow for a specified number and distribution of slow control rods, a single stuck control rod and an assumed single failure. Therefore, if all control rods met the scram time limits found in ITS Table 3.1.4-1, the analytical scram reactivity assumptions are satisfied. If control rods do not meet the scram time limits, ITS LCO 3.1.4 specifies the number and distribution of these "slow" control rods to ensure the analytical scram reactivity assumptions are still satisfied. If the number of slow rods is more than 8 or the rods do not meet the separation requirements, the unit must be shutdown. This change is designated as more restrictive because explicit requirements have been included in the Technical Specifications to cover conditions not currently addressed in the CTS. ITS 3.1.4 specifies limitations on scram times for each individual control rod. That is,

DISCUSSION OF CHANGES
ITS 3.1.4, CONTROL ROD SCRAM TIMES

currently, the "average time" of all rods or a group can be improved by a few fast scramming rods, even when there may be more than 8 slow rods, as defined in the proposed Specification. Therefore, ITS 3.1.4 limits the number of slow rods to 8 and ensures no more than 2 slow rods occupy adjacent locations. The maximum scram time requirement has been added to the ITS (see Discussion of Changes for ITS 3.1.3) for the purpose of defining the threshold between a slow control rod and an inoperable control rod even though the analyses to determine the LCO scram time limits assumed slow control rods did not scram. Note 2 to ITS Table 3.1.4-1 ensures that a control rod is not inadvertently considered "slow" when the scram time exceeds 7 seconds. This change is designated as more restrictive because explicit requirements have been included in the Technical Specifications to cover conditions not currently addressed in the CTS.

M.2 CTS 4.3.C requires each OPERABLE rod to be scram time tested during each operating cycle, however, it also states that if testing is not accomplished during reactor power operation, the measured scram time may be extrapolated to the reactor power operation condition. ITS SR 3.1.4.1 requires verification that each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig prior to exceeding 40% RTP after each reactor shutdown ≥ 120 days. ITS SR 3.1.4.2 requires verification that, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig every 200 days of cumulative operation in MODE 1. ITS SR 3.1.4.3 requires verification that each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure prior to declaring a control rod OPERABLE after work on control rod or CRD System that could affect scram time. ITS SR 3.1.4.4 requires verification that each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 800 psig prior to exceeding 40% RTP after fuel movement within the affected core cell and prior to exceeding 40% RTP after work on a control rod or the CRD System that could affect the scram time. In addition, a Surveillance Note has been added that states "During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator." This changes the CTS by requiring a scram time test to be performed prior to declaring a control rod OPERABLE after work on control rod or CRD System that could affect scram time. It also requires the unit to complete scram time testing of affected control rods prior to exceeding 40% RTP after fuel movement within the affected cell and after work on a control rod or the CRD System that could affect the scram time. In addition, if the reactor is shutdown for ≥ 120 days, a scram time test of each control rod is required to be performed prior to exceeding 40% RTP, and, after every 200 days of cumulative operation in MODE 1, a representative sample of control rods must be scram time tested. Finally the change requires the single control rod scram time Surveillance to be performed with the CRD pumps isolated form the associated scram accumulator.

The purpose of CTS 4.3.C is to ensure the control rods can meet the scram reactivity insertion requirements to support the unit safety analysis. This change provides more explicit control rod scram time testing requirements than in the CTS to ensure control rods are OPERABLE prior to entering MODE 2 when work has been performed on the control rod or CRD System that could affect its scram time. Soon after entering MODE 2 (and prior to exceeding 40% RTP), scram

DISCUSSION OF CHANGES
ITS 3.1.4, CONTROL ROD SCRAM TIMES

time tests are required at steam dome pressures of ≥ 800 psig (after fuel movement within the affected cell and after work on control rod or CRD System that could affect scram time) to confirm the scram time performance at the most limiting conditions. In addition, if the reactor has been shutdown for ≥ 120 days, each control rod must be tested prior to exceeding 40% RTP. Furthermore, every 200 days of cumulative operation in MODE 1, a representative sample of control rods must be scram time tested to ensure the limits of Table 3.1.4-1 are met. The scram time criteria at < 800 psig will be lower than the scram time values specified in Table 3.1.4-1 for ≥ 800 psig. The criterion is established based on previously determined correlations. Satisfying the test at these conditions (< 800 psig) will almost guarantee with a high probability the acceptance criteria at ≥ 800 psig will be satisfied. This low pressure testing is required when work has been performed on a control rod or CRD System that could affect scram time. Affected control rods cannot be declared OPERABLE until this test is performed. The test at ≥ 800 psig is necessary because at approximately 800 psig there are competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure ≥ 800 psig ensures that the measured scram times will be within the specified limits at higher reactor pressures. The first Frequency of ITS SR 3.1.4.4 is prior to exceeding 40% RTP after fuel movement within the affected core cell. This Frequency will basically require a high pressure scram test for each control rod after a refueling outage. CTS 4.3.C requires a test during each Operating Cycle, which is defined in CTS 1.0.N as the interval between the end of one refueling outage and the end of the next subsequent refueling outage. This Surveillance Frequency in ITS SR 3.1.4.4 will ensure that the necessary scram testing is performed shortly after MODE 2 is entered after a refueling outage (i.e., prior to exceeding 40% RTP). This Frequency is necessary since control rod scram performance is necessary in establishing MCPR operating limits and to ensure the MCPR Safety Limit is met during a unit transient. The second Frequency of ITS SR 3.1.4.4 is prior to exceeding 40% RTP after work on a control rod or CRD System that could affect scram time. This Frequency will basically only require a high pressure scram test for each affected control rod after a non-refueling outage if work has been performed on the control rod or CRD System. Specific examples of work that could affect the scram times are (but are not limited to) the following: removal of any CRD for maintenance or modification; replacement of a control rod; and maintenance or modification of a scram solenoid pilot valve, scram valve, accumulator, isolation valve or check valve in the piping required for scram. ITS SR 3.1.4.1 has been added to ensure that if the unit has been shutdown for a long period of time (≥ 120 days), the control rods are scram timed to ensure compliance with the acceptance criteria. This is necessary to ensure that any maintenance activity over this shutdown period has not affected the control rod scram capabilities and due to the fact that the control rods are normally not exercised during shutdown conditions. ITS SR 3.1.4.2 has been added to ensure a representative sample of control rods are periodically tested (every 200 days of cumulative operation in MODE 1) to ensure the scram times remain within the limits of Table 3.1.4-1 throughout the cycle. The four SRs of this LCO are modified by a Note stating that during a single control rod scram time Surveillance, the CRD pumps shall be isolated from the associated scram accumulator. With the CRD pump isolated (i.e., charging valve closed), the influence of the CRD pump head will affect the single control rod scram times.

DISCUSSION OF CHANGES
ITS 3.1.4, CONTROL ROD SCRAM TIMES

This Note restriction is not necessary during a full core scram since the CRD pump head would be seen by all control rods and would have a negligible effect on the scram insertion times. This change is designated as more restrictive because the Surveillances prescribe more frequent Surveillance Frequencies than are required by the CTS.

M.3 CTS 3.3.G.1 requires the unit to be in cold shutdown (MODE 4) within 24 hours if CTS 3.3.C is not met. ITS 3.1.4 ACTION A requires the unit to be in MODE 3 in 12 hours when ITS LCO 3.1.4 is not met. This changes the CTS by requiring the unit to be in MODE 3 in 12 hours instead of 24 hours. The change to the unit condition required to be achieved (MODE 3 versus MODE 4) is discussed in DOC A.2.

The purpose of CTS 3.3.G.1 is to place the unit in a condition where the scram time limits of the control rods are not required. This change is acceptable because the allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. The requirement to be in MODE 3 in 12 hours is designated as more restrictive since MODE 3 must be achieved in a faster time limit than is currently required.

M.4 CTS 3.3.C.1 requires the scram times to be within the limits in the "reactor power operation condition." ITS LCO 3.1.4 is Applicable in MODES 1 and 2. This changes the CTS by requiring the scram time limits to be met in MODE 2 ≤ 1% RATED THERMAL POWER (RTP).

The purpose of CTS 3.3.C.1 is to ensure the negative scram reactivity is consistent with those values assumed in the accident and transient analysis. This change expands the Applicability to require the scram time limits to be met at all times when in MODE 2, instead of when > 1% RTP (the CTS 1.0.O definition states that Power Operation is when reactor power is > 1% RTP). This change is acceptable since the scram time limits must be met in MODE 2 because the reactor is critical or control rods are withdrawn (thus the need for the control rods to be capable of properly scramming in the assumed time exists). This change is designated as more restrictive because the LCO will be applicable under more reactor conditions.

<u>RELOCATED SPECIFICATIONS</u>

None

<u>REMOVED DETAIL CHANGES</u>

None

DISCUSSION OF CHANGES
ITS 3.1.4, CONTROL ROD SCRAM TIMES

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

3.3.C.1, 3.3.C.2 LCO 3.1.4

a. No more than [10] 8 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1, and 1

b. No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations.

3.3.C.1 APPLICABILITY: MODES 1 and 2.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.3.G.1	A. Requirements of the LCO not met.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----

DOC M.2 During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

	SURVEILLANCE		FREQUENCY	
4.3.C	SR 3.1.4.1	Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ [800] psig.	Prior to exceeding 40% RTP after each reactor shutdown ≥ 120 days	1
4.3.C	SR 3.1.4.2	Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ [800] psig.	[200] 120 days cumulative operation in MODE 1	TSTF-460 1 1

CTS

SURVEILLANCE REQUIREMENTS (continued)

CTS	SURVEILLANCE		FREQUENCY
4.3.C	SR 3.1.4.3	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with any reactor steam dome pressure.	Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time
4.3.C	SR 3.1.4.4	Verify each affected control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ [800] psig.	Prior to exceeding 40% RTP after fuel movement within the affected core cell 1 AND Prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

CTS

NOTES

DOC M.1 1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

DOC M.1 2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position [06]. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow." (1)

3.3.C.1, 3.3.C.2 NOTCH POSITION	SCRAM TIMES[(a)(b)] (seconds) WHEN REACTOR STEAM DOME PRESSURE ≥ [800] psig (1)
[46]	[0.44]
[36]	[1.08]
[26]	[1.83]
[06]	[3.35]

(1)

3.3.C.1, 4.3.C (a) Maximum scram time from fully withdrawn position[,] based on de-energization of scram pilot valve solenoids at time zero. (2)

DOC M.1 (b) Scram times as a function of reactor steam dome pressure[,] when < 800 psig are within established limits. (2)

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.4, CONTROL ROD SCRAM TIMES

1. The brackets are removed and the proper plant specific information/value is provided.

2. Typographical/grammatical error corrected.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Scram Times

INSERT 1

BASES

BACKGROUND

The scram function of the Control Rod Drive (CRD) System ~~controls reactivity changes~~ during abnormal ~~operational transients~~ to ensure that ~~specified acceptable fuel design limits are not exceeded~~ (Ref. 1). The control rods are scrammed by positive means using hydraulic pressure exerted on the CRD piston. 1

When a scram signal is initiated, control air is vented from the scram valves, allowing them to open by spring action. Opening the exhaust valve reduces the pressure above the main drive piston to atmospheric pressure, and opening the inlet valve applies the accumulator or reactor pressure to the bottom of the piston. Since the notches in the index tube are tapered on the lower edge, the collet fingers are forced open by cam action, allowing the index tube to move upward without restriction because of the high differential pressure across the piston. As the drive moves upward and the accumulator pressure reduces below the reactor pressure, a ball check valve opens, letting the reactor pressure complete the scram action. If the reactor pressure is low, such as during startup, the accumulator will fully insert the control rod in the required time without assistance from reactor pressure.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 2, 3, and 4. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. The resulting negative scram reactivity forms the basis for the determination of plant thermal limits (e.g., the MCPR). Other distributions of scram times (e.g., several control rods scramming slower than the average time with several control rods scramming faster than the average time) can also provide sufficient scram reactivity. Surveillance of each individual control rod's scram time ensures the scram reactivity assumed in the DBA and transient analyses can be met.

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), which , ensure that no fuel damage will occur if these limits are not exceeded. 1
At ≥ ~~Above~~ 800 psig, the scram function is designed to insert negative 2

and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"

1

INSERT 1

is designed to accommodate plant operational transients or maneuvers which might be expected without compromising safety and without fuel damage

BASES

APPLICABLE SAFETY ANALYSES (continued)

reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL, during the analyzed limiting power transient. Below 800 psig, the scram function is assumed to perform design during the control rod drop accident (Ref. 5) and, therefore, also provides protection against violating fuel ~~damage~~ limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis, the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits. 1

Control rod scram times satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The scram times specified in Table 3.1.4-1 ~~(in the accompanying LCO)~~ are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met (Ref. 6). To account for single failures and "slow" scramming control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin that allows up to approximately 7% of the control rods (e.g., ~~137~~ 121 x 7% ~~= 10~~ = 8) to have scram times exceeding the specified limits (i.e., "slow" control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished through measurement of the "dropout" times. To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed "slow" control rods may occupy adjacent locations (face or diagonal). 2

Table 3.1.4-1 is modified by two Notes which state that control rods with scram times not within the limits of the Table are considered "slow" and that control rods with scram times > 7 seconds are considered inoperable as required by SR 3.1.3.4.

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed (LCO 3.1.3). Slow scramming control rods may be conservatively declared inoperable and not accounted for as "slow" control rods.

BASES

APPLICABILITY

In MODES 1 and 2, a scram is assumed to function during transients and accidents analyzed for these plant conditions. These events are assumed to occur during startup and power operation; therefore, the scram function of the control rods is required during these MODES. In MODES 3 and 4, the control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram capability during these conditions. Scram requirements in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."

ACTIONS

A.1

When the requirements of this LCO are not met, the rate of negative reactivity insertion during a scram may not be within the assumptions of the safety analyses. Therefore, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The four SRs of this LCO are modified by a Note stating that during a single control rod scram time surveillance, the CRD pumps shall be isolated from the associated scram accumulator. With the CRD pump isolated, (i.e., charging valve closed) the influence of the CRD pump head does not affect the single control rod scram times. During a full core scram, the CRD pump head would be seen by all control rods and would have a negligible effect on the scram insertion times.

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on an assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 800 psig demonstrates acceptable scram times for the transients analyzed in References ~~3 and 4~~ 2 and 3. 1

Maximum scram insertion times occur at a reactor steam dome pressure of approximately 800 psig because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure ≥ 800 psig ensures that the measured scram times will be within the

BASES

SURVEILLANCE REQUIREMENTS (continued)

specified limits at higher pressures. Limits are specified as a function of reactor pressure to account for the sensitivity of the scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed. To ensure that scram time testing is performed within a reasonable time following a shutdown ≥ 120 days or longer, control rods are required to be tested before exceeding 40% RTP following the shutdown. This Frequency is acceptable considering the additional surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected by fuel movement within the associated core cell and by work on control rods or the CRD System. 3

SR 3.1.4.2

Additional testing of a sample of control rods is required to verify the continued performance of the scram function during the cycle. A representative sample contains at least 10% of the control rods. The sample remains representative if no more than 20% of the control rods in the sample tested are determined to be "slow." With more than 20% of the sample declared to be "slow" per the criteria in Table 3.1.4-1, additional control rods are tested until this 20% criterion (e.g., 20% of the entire sample size) is satisfied, or until the total number of "slow" control rods (throughout the core, from all surveillances) exceeds the LCO limit. For planned testing, the control rods selected for the sample should be different for each test. Data from inadvertent scrams should be used whenever possible to avoid unnecessary testing at power, even if the control rods with data may have been previously tested in a sample. The 120 day Frequency is based on operating experience that has shown control rod scram times do not significantly change over an operating cycle. This Frequency is also reasonable based on the additional Surveillances done on the CRDs at more frequent intervals in accordance with LCO 3.1.3 and LCO 3.1.5, "Control Rod Scram Accumulators." 7.5 TSTF-460 3 200 TSTF-460 4

SR 3.1.4.3

When work that could affect the scram insertion time is performed on a control rod or the CRD System, testing must be done to demonstrate that each affected control rod retains adequate scram performance over the range of applicable reactor pressures ~~from zero to the maximum~~ . 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

scram time

found in the Technical Requirements Manual (Ref. 7) and are

~~permissible pressure.~~ The scram testing must be performed once before declaring the control rod OPERABLE. The required scram time testing must demonstrate the affected control rod is still within acceptable limits. The limits for reactor pressures < 800 psig are established based on a high probability of meeting the acceptance criteria at reactor pressures ≥ 800 psig. Limits for ≥ 800 psig are found in Table 3.1.4-1. If testing demonstrates the affected control rod does not meet these limits, but is within the 7-second limit of Table 3.1.4-1, Note 2, the control rod can be declared OPERABLE and "slow." 2 1

Specific examples of work that could affect the scram times are (but are not limited to) the following: removal of any CRD for maintenance or modification; replacement of a control rod; and maintenance or modification of a scram solenoid pilot valve, scram valve, accumulator, isolation valve or check valve in the piping required for scram.

The Frequency of once prior to declaring the affected control rod OPERABLE is acceptable because of the capability to test the control rod over a range of operating conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

SR 3.1.4.4

When work that could affect the scram insertion time is performed on a control rod or CRD System, or when fuel movement within the reactor pressure vessel occurs, testing must be done to demonstrate each affected control rod is still within the limits of Table 3.1.4-1 with the reactor steam dome pressure ≥ 800 psig. Where work has been performed at high reactor pressure, the requirements of SR 3.1.4.3 and SR 3.1.4.4 can be satisfied with one test. For a control rod affected by work performed while shut down, however, a zero pressure and high pressure test may be required. This testing ensures that, prior to withdrawing the control rod for continued operation, the control rod scram performance is acceptable for operating reactor pressure conditions. Alternatively, a control rod scram test during hydrostatic pressure testing could also satisfy both criteria. When fuel movement within the reactor pressure vessel occurs, only those control rods associated with the core cells affected by the fuel movement are required to be scram time tested. During a routine refueling outage, it is expected that all control rods will be affected.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of once prior to exceeding 40% RTP is acceptable because of the capability to test the control rod over a range of operating conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10. USAR, Section 1.2.2 1
2. FSAR, Section [4.2.3.2.2.4]. Chapter 14
3. FSAR, Section [5A.4.3]. Chapter 14A 4
4. FSAR, Section [15.1]. Chapter 3

1 U

5. NEDE-24011-P-A-9, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, September 1988. 1

(revision as specified in Specification 5.6.3)

6. Letter from R.F. Janecek (BWROG) to R.W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, September 17, 1987.

7. Technical Requirements Manual. 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.4 BASES, CONTROL ROD SCRAM TIMES

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. Typographical/grammatical error corrected.

4. The brackets have been removed and the proper plant specific information has been provided.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.4, CONTROL ROD SCRAM TIMES

There are no specific NSHC discussions for this Specification.

ATTACHMENT 5

ITS 3.1.5, Control Rod Scram Accumulators

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS | ITS

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ITS	3.0 LIMITING CONDITIONS FOR OPERATION	ITS	4.0 SURVEILLANCE REQUIREMENTS
	D. Control Rod Accumulators		D. Control Rod Accumulators
LCO 3.1.5	Applicability: MODES 1 and 2 — Control rod accumulators (See ITS 3.9.5) shall be operable ~~in the Startup, Run, or Refuel modes~~ except as provided below.	SR 3.1.5.1	Once per ~~12 hours~~ 7 days check the status ~~in the control room~~ of the required Operable accumulator pressure ~~and level alarms~~ is ≥ 940 psig.
	If a control rod with an inoperable accumulator is inserted "full-in" and ~~either its directional control valves are electrically~~ disarmed ~~or it is hydraulically isolated~~, it shall not be considered to have an inoperable accumulator.		Add proposed ACTIONS Note
ACTIONS A, B, and C	1. ~~In the Startup or Run Mode,~~ a rod accumulator may be inoperable provided that no other control rod within two control rod cells in any direction has a: (a) Inoperable accumulator, or (b) Directional control valve electrically disarmed while in a non-fully inserted position.		Add proposed ACTIONS A, B, and C

L.1 M.1 L.2 M.1 A.2 LA.1 L.3 L.3

A.1

ITS

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3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
F. Scram Discharge Volume	F. Scram Discharge Volume
1. During reactor operation, the scram discharge volume vent and drain valves shall be operable, except as specified below.	The scram discharge volume vent and drain valves shall be cycled quarterly.
2. If any scam discharge volume vent or drain valve is made or found inoperable, the integrity of the scram discharge volume shall be maintained by either:	Once per operating cycle verify the scram discharge volume vent and drain valves close within 30 seconds after receipt of a reactor scram signal and open when the scram is reset.
a. Verifying daily, for a period not to exceed 7 days, the operability of the redundant valve(s), or	
b. Maintaining the inoperable valve(s), or the associated redundant valve(s), in the closed position. Periodically the inoperable and the redundant valve(s) may both be in the open position to allow draining the scram discharge volume.	
If a or b above cannot be met, at least all but one operable control rods (not including rods removed per specification 3.10.E or inoperable rods allowed by 3.3.A.2) shall be fully inserted within ten hours.	
G. Required Action	
If Specifications 3.3.~~A through D~~ above are not met, an orderly shutdown ~~shall be initiated and have reactor in the cold shutdown condition~~ within 24 hours.	

See ITS 3.1.8

A.3

1.

ACTION D

(except when the reactor mode switch is in the Refuel position)

A.3

Add proposed ACTION D

L.3

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DISCUSSION OF CHANGES
ITS 3.1.5, CONTROL ROD SCRAM ACCUMULATORS

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.3.D states, in part, that if "a" control rod with an inoperable accumulator is inserted full-in and either its directional control valves are electrically disarmed or is hydraulically isolated, it shall not be considered to have an inoperable accumulator. CTS 3.3.D.1 states, in part, that "a" rod accumulator may be inoperable provided that no other control rod within two control rod cells in any direction has an inoperable accumulator or directional control valve are electrically disarmed while in a non-fully inserted position. These CTS Actions do not limit the number of accumulators to which these Actions apply. ITS 3.1.5 ACTIONS Note allows separate Condition entry for each control rod scram accumulator. This changes the CTS by adding an explicit Note for separate Condition entry for each control rod scram accumulator.

The purpose of CTS 3.3.D and CTS 3.3.D.1, in part, is to provide compensatory actions for an inoperable scram accumulator on an individual basis. ITS 3.1.5 ACTION Note "Separate Condition entry is allowed for each control rod scram accumulator" has been added and provides more explicit instructions for proper application of the ACTIONS for Technical Specifications compliance. In conjunction with proposed Specification 1.3, "Completion Times," this Note provides direction consistent with the intent of the existing CTS ACTIONS for inoperable control rod accumulators. Upon discovery of each inoperable accumulator, each specified ACTION is applied, regardless of previous application to other inoperable accumulators. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 These changes to CTS 3.3.G are provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the USNRC for approval in NMC letter L-MT-05-013, from Thomas J. Palmisano (NMC) to USNRC, dated April 12, 2005. As such, these changes are administrative.

MORE RESTRICTIVE CHANGES

M.1 CTS 4.3.D requires a check of the accumulator pressure alarm located in the control room. ITS SR 3.1.5.1 requires a verification that each control rod scram accumulator pressure is $\geq$ 940 psig. This changes the CTS by providing an explicit value for control rod accumulator pressure, in lieu of specifying the alarm in the control room must be checked.

The purpose of CTS 4.3.D is to ensure that each control rod scram accumulator is OPERABLE. ITS SR 3.1.5.1 includes the acceptance criteria for accumulator

DISCUSSION OF CHANGES
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pressure (≥ 940 psig) consistent with current Monticello practice, and requires verification that each accumulator meets this pressure criterion. Although this change is consistent with current practice, adding this acceptance criterion and verification requirement in ITS SR 3.1.5.1 is an additional restriction on unit operation since control of this requirement will now be governed by Technical Specifications. This change is designated as more restrictive because it adds an explicit Surveillance limit that does not appear in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.3.D states, in part, that if a control rod with an inoperable accumulator is inserted full-in and either its "directional control valves are electrically "disarmed" or it is hydraulically isolated," it shall not be considered to have an inoperable accumulator. ITS 3.1.3 ACTION C covers the compensatory actions for one or more inoperable control rods (control rods inoperable as a result of an inoperable accumulator is covered by this condition when declared inoperable) ITS 3.1.3 Required Action C.2 states "Disarm the associated CRD," but does not provide the specific details of how to disarm the CRD. This changes the CTS by relocating the details that the "directional control valves are electrically" disarmed "or it is hydraulically isolated" to the ITS Bases.

The removal of these details for performing Required Actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 3.1.3 Required Action C.2 still retains the requirement to disarm the CRD. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 (Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change) CTS 4.3.D requires a check of the status in the control room of the required OPERABLE accumulator every 12 hours. ITS SR 3.1.5.1 requires a similar verification that the pressure in each accumulator is ≥ 940 psig every 7 days. This changes the CTS extending the Surveillance Frequency from once every 12 hours to every 7 days.

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ITS 3.1.5, CONTROL ROD SCRAM ACCUMULATORS

The purpose of CTS 4.3.D is to ensure the control rod scram accumulators are OPERABLE to support the associated control rod scram function. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. This change allows the unit to perform the Surveillance every 7 days instead of every 12 hours. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications (i.e., alarm) available in the control room. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.2 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS 4.3.D requires, in part, the check of the status in the control room of the required OPERABLE accumulator level alarm. The ITS does not include this requirement. This changes the CTS by deleting the requirement to verify the alarm for accumulator level in the control room.

The purpose of CTS 4.3.D is to ensure each control rod scram accumulator is OPERABLE to support the associated control rod scram function. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. ITS SR 3.1.5.1 requires verification that the accumulator pressure is within the pressure limit for each accumulator. The actual limit has been added as described in DOC M.1. This change deletes the requirement to verify OPERABILITY of the control rod accumulators via the accumulator level alarm in the control room. The ISTS do not specify OPERABILITY requirements for equipment that only provides indication to support OPERABILITY of a system or component. The control rod scram accumulator level alarm does not necessarily relate directly to accumulator OPERABILITY. Control of the availability of, and necessary compensatory activities, for alarms, are addressed by unit procedures and policies. The requirement to verify control rod scram accumulator pressure (which does relate directly to accumulator OPERABILITY) is within limits is still maintained in ITS SR 3.1.5.1. Therefore, the requirements associated with the control rod accumulator level alarm are proposed to be removed from the Technical Specifications. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L.3 *(Category 4 – Relaxation of Required Action)* CTS 3.3.D states, in part, that if a control rod with an inoperable accumulator is inserted full-in and is disarmed, it shall not be considered to have an inoperable accumulator. CTS 3.3.D.1 also states a control rod scram accumulator may be inoperable provided that no other control rod within two control rod cells in any direction has an inoperable accumulator or a directional control valve electrically disarmed while in a non-fully inserted position. CTS 3.3.G.1 states, in part, that if Specification 3.3.D is not met, an orderly shutdown shall be initiated and the reactor shall be placed in the cold shutdown (MODE 4) condition within 24 hours. CTS 3.3.D and CTS 3.3.D.1 do not provide any time to insert control rods associated with inoperable control rod accumulators, therefore as soon as it is determined that a control rod accumulator is inoperable and the provisions of CTS 3.3.D.1 are not met, CTS 3.3.G.1 must be immediately entered. ITS 3.1.5 ACTION A covers the condition of one control rod scram accumulator inoperable with reactor steam dome pressure ≥ 900 psig, and requires the declaration within 8 hours that either the associated control rod scram time is slow (ITS 3.1.5 Required Action A.1) or the associated control rod is inoperable (ITS 3.1.5 Required Action A.2). The requirement to declare the associated control rod slow is only applicable if the associated control rod scram time was within limits during the last scram time Surveillance. ITS 3.1.5 ACTION B covers the Condition for two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 900 psig, and requires the restoration of charging water header pressure to ≥ 940 psig within 20 minutes from discovery of Condition B (i.e., two or more control rod scram accumulators inoperable with steam dome pressure ≥ 900 psig) concurrent with charging water header pressure < 940 psig (ITS 3.1.5 Required Action B.1) and within 1 hour to either declare the associated control rod scram time slow (ITS 3.1.5 Required Action B.2.1) or declare the associated control rod inoperable (ITS 3.1.5 Required Action B.2.2). The requirement to declare the associated control rod scram time slow is only applicable if the associated control rod scram time was within limits during the last scram time Surveillance. ITS 3.1.5 ACTION C covers the condition for one or more control rod scram accumulators inoperable with reactor steam dome pressure < 900 psig, and requires the immediate verification that all control rods associated with inoperable accumulators are fully inserted upon discovery of charging water header pressure < 940 psig (ITS 3.1.5 Required Action C.1) and the declaration within 1 hour that the associated control rod is inoperable (ITS 3.1.5 Required Action C.2). ITS 3.1.5 ACTION D covers the condition when Required Action B.1 or C.1 and associated Completion Time is not met, and requires the immediate placement of the reactor mode switch in the shutdown position (Required Action D.1). This Required Action is not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. This changes the CTS in several ways, some administrative, some more restrictive, and some less restrictive. However, all these changes are discussed in this single less restrictive change discussion for clarity. The individual changes and their justification and categorization are as follows:

- The ITS 3.1.5 ACTIONS for inoperable control rods are configured based upon the reactor steam dome pressure (i.e., ≥ 900 psig and < 900 psig). At reactor pressures < 900 psig and with control rod scram accumulators inoperable, the resultant control rod scram time is not expected to satisfy the minimum scram time requirement of ITS SR 3.1.3.4 (i.e., 7 second scram time requirement). ITS 3.1.5 ACTION C covers the condition of one or more control rod scram accumulators inoperable with reactor steam dome pressure < 900 psig. When ACTION C is entered and it is discovered that charging water header pressure is < 940 psig, an immediate verification is required to ensure that all control rods associated with inoperable scram accumulators are fully inserted. With one or more control rod scram accumulators inoperable and the reactor steam dome pressure < 900 psig, the pressure supplied to the charging water header must be adequate to ensure that accumulators remain charged. With the reactor steam dome pressure < 900 psig, the function of the accumulators in providing the scram force becomes much more important since the scram function could become

DISCUSSION OF CHANGES
ITS 3.1.5, CONTROL ROD SCRAM ACCUMULATORS

severely degraded during a depressurization event or at low reactor pressures. Therefore, immediately upon discovery of charging water header pressure < 940 psig, concurrent with Condition C, all control rods associated with inoperable accumulators must be verified to be fully inserted. If ITS 3.1.5 Required Action C.1 and its associated Completion Time cannot be met ITS 3.1.4 ACTION D requires the immediate placement of the reactor mode switch to the shutdown position. This will ensure all control rods are inserted into the reactor core and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. In addition, within 1 hour the associated control rod must be declared inoperable. Withdrawn control rods with inoperable accumulators may fail to scram under these low pressure conditions. When a control rod is declared inoperable, ITS 3.1.3 ACTION C requires the insertion of the control rod within 3 hours and the disarming of the control rod drive in 4 hours. CTS 3.3.D does not provide any additional restrictions for inoperable control rod scram accumulators at low reactor steam dome pressures. Thus, CTS allows rods to remain not-fully inserted at low reactor pressures (< 900 psig) with inoperable accumulators. These changes are acceptable because they place the control rods in a safer condition under the specified conditions. This change is more restrictive because ITS 3.1.5 ACTION C requires the control rod to be declared inoperable and ITS 3.1.3 ACTION C will require the rod to be inserted and the control rod drive mechanism disarmed whereas in the CTS the control rod may remain withdrawn as long as the criteria in CTS 3.3.D.1.(a) and (b) are met. In addition, it is also more restrictive as a result of the addition of the explicit requirements for when charging water header pressure is not within limit and the requirement to place the reactor mode switch in the shutdown condition under certain conditions.

- The ITS 3.1.5 ACTIONS for inoperable control rods are configured based on the reactor steam dome pressure (i.e., ≥ 900 psig and < 900 psig). At reactor pressures ≥ 900 psig and with a control rod scram accumulators inoperable, the resultant scram time of the associated rod may still satisfy the minimum scram time requirements of ITS SR 3.1.3.4 (i.e., 7 second scram time requirement) and the scram time criteria of ITS Table 3.1.4-1. Therefore, ITS 3.1.5 ACTION A allows the associated control rods to be declared either inoperable or slow. This declaration must be performed within 8 hours. If two or more control rod scram accumulators are inoperable with reactor steam dome pressure ≥ 900 psig, ITS 3.1.5 ACTION B requires the associated control rods to also be declared either inoperable or slow. However, this declaration must be performed within 1 hour. If during this condition (i.e., in Condition B), it is found that the charging water header pressure is < 940 psig, it must be restored to ≥ 940 psig within 20 minutes. If this cannot be met the reactor mode switch must be placed in the shutdown position. If a control rod has an inoperable accumulator in the CTS, it must be inserted and disarmed or it may be allowed to remain withdrawn as long as the criteria of CTS 3.3.D.1.(a) and (b) are met. The CTS essentially allows 24 hours to satisfy the requirements of CTS 3.3.D or 3.3.D.1, since CTS 3.3.G.1 allows 24 hours to place the reactor in cold shutdown. In the ITS, 8 hours is allowed to declare the rod inoperable or slow if one control rod scram accumulator is inoperable with reactor steam dome pressure ≥ 900 psig. If declared inoperable, ITS 3.1.3 ACTION C allows 3 hours to fully insert the rod and

DISCUSSION OF CHANGES
ITS 3.1.5, CONTROL ROD SCRAM ACCUMULATORS

4 hours to disarm it. Therefore, with one control rod scram accumulator inoperable with reactor steam dome pressure ≥ 900 psig, the ITS requires the same operations to be completed in 12 hours (8 hours to declare the control rod inoperable and 4 hours to insert and disarm it). With one or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 900 psig, the ITS requires the same operations to be completed in 5 hours (1 hour to declare the control rod inoperable and 4 hours to insert and disarm it). This change is acceptable since it places the reactor in a safer condition sooner under the same conditions and prescribes explicit requirements for when the charging water header pressure requirements cannot be met. This portion of the change is more restrictive since less time is provided to perform the same actions (insert and disarm control rods) and provides more explicit action for when charging water header pressure is not within limits.

- CTS 3.3.D.1 includes a provision that allows control rods with inoperable control rod scram accumulators to remain withdrawn as long as there is no other inoperable control rod (i.e., inoperable accumulator, or directional control valve electrically disarmed while in a non-fully inserted position) within two control rod cells in any direction. ITS 3.1.5 ACTION A includes a similar allowance only if the control rod is declared slow. When a rod is declared slow, an evaluation is normally performed to ensure LCO 3.1.4 is met. ITS LCO 3.1.4.a includes a requirement that limits the total number of OPERABLE control rods that are "slow" to 8 and ITS LCO 3.1.4.b includes a requirement that allows no more than 2 OPERABLE control rods that are slow to occupy adjacent locations. This changes the CTS by effectively limiting the total number of withdrawn control rods with inoperable control rod scram accumulators to 8. This also changes the CTS by allowing 2 OPERABLE control rods that are slow (i.e., the control rod has an inoperable accumulator and is not fully inserted) to occupy adjacent locations and allows other slow control rods to only be separated by a single OPERABLE control rod. This change is acceptable since an analysis has been performed to ensure the control rod scram reactivity can be met when in this configuration. The scram times specified in ITS Table 3.1.4-1 are required to ensure that the scram reactivity assumed in the design basis accident and transient analyses is met. To account for single failures and slow scramming control rods (control rods with inoperable scram accumulators), the scram times specified in ITS Table 3.1.4-1 are faster than those assumed in the design basis analyses. The scram times have a margin that allows up to approximately 7% of the control rods (i.e., 8) to have scram times exceeding the specified limits (i.e., slow control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed slow control rods may occupy adjacent locations. This change is acceptable because the limitations placed on the inoperable control rod scram accumulators will ensure the safety analyses will be met. The change limiting the total number of slow control rods to 8 is more restrictive than the CTS while the changes related to separation criteria are less restrictive than the CTS.

DISCUSSION OF CHANGES
ITS 3.1.5, CONTROL ROD SCRAM ACCUMULATORS

- CTS 3.3.G.1 states, in part, that if Specification 3.3.D is not met, an orderly shutdown shall be initiated and the reactor shall be placed in the cold shutdown (MODE 4) condition within 24 hours. ITS 3.1.5 ACTION D covers the condition when Required Action B.1 or C.1 and associated Completion Time is not met, and requires the immediate placement of the reactor mode switch in the shutdown position. Placing the reactor mode switch in shutdown places the reactor in hot shutdown (MODE 3). This change is considered acceptable since CTS 3.3.D, in part, is applicable in the Startup and Run conditions, i.e., MODES 1 and 2. Thus, once MODE 3 is achieved, continuation to MODE 4 is no longer required. Therefore, this change is considered administrative since the technical requirements are not being changed; the change is being made to ensure the shutdown actions are consistent with the current Applicability.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

3.3.D LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----

DOC A.2 Separate Condition entry is allowed for each control rod scram accumulator.

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.3.D, 3.3.D.1	A. One control rod scram accumulator inoperable with reactor steam dome pressure ≥ [900] psig.	A.1 -----NOTE----- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. ----------		1
		Declare the associated control rod scram time "slow."	8 hours	
		OR		
		A.2 Declare the associated control rod inoperable.	8 hours	
3.3.D, 3.3.D.1	B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ [900] psig.	B.1 Restore charging water header pressure to ≥ [940] psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < [940] psig	1
		AND		

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.3.D, 3.3.D.1		B.2.1 ----------NOTE---------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. ------------------------------		
		Declare the associated control rod scram time "slow."	1 hour	
		OR		3
		B.2.2 Declare the associated control rod inoperable.	1 hour	
3.3.D, 3.3.D.1	C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < [900] psig.	C.1 Verify all control rods associated with inoperable accumulators are fully inserted.	Immediately upon discovery of charging water header pressure < [940] psig	1
		AND		
		C.2 Declare the associated control rod inoperable.	1 hour	
3.3.G.1	D. Required Action and associated Completion Time ~~of Required Action B.1 or C.1~~ not met.	D.1 ----------NOTE---------- Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. ------------------------------		2
		Place the reactor mode switch in the shutdown position.	Immediately	

CTS

SURVEILLANCE REQUIREMENTS

CTS		SURVEILLANCE	FREQUENCY	
4.3.D	SR 3.1.5.1	Verify each control rod scram accumulator pressure is ≥ [940] psig.	7 days	1

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.5, CONTROL ROD SCRAM CONTROL ROD SCRAM ACCUMULATORS

1. The brackets are removed and the proper plant specific information/value is provided.

2. The wording of the Condition has been made to be consistent with a similar type of requirement in another Specification in NUREG-1433, Rev. 3 (i.e., ISTS 3.5.2 Condition D). This change was also approved in the ITS conversion for the four most recently approved BWR conversions (Quad Cities 1 and 2, Dresden 2 and 3, LaSalle 1 and 2, and FitzPatrick).

3. Typographical error corrected.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Rod Scram Accumulators

BASES

BACKGROUND	The control rod scram accumulators are part of the Control Rod Drive (CRD) System and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen, which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.1.4, "Control Rod Scram Times."
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 1, 2, and 3. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. OPERABILITY of each individual control rod scram accumulator, along with LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.4, ensures that the scram reactivity assumed in the DBA and transient analyses can be met. The existence of an inoperable accumulator may invalidate prior scram time measurements for the associated control rod. The scram function of the CRD System, and therefore the OPERABILITY of the accumulators, protects the MCPR Safety Limit (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded (see Bases for LCO 3.1.4). In addition, the scram function at low reactor vessel pressure (i.e., startup conditions) provides protection against violating fuel design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). Control rod scram accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The OPERABILITY of the control rod scram accumulators is required to ensure that adequate scram insertion capability exists when needed over the entire range of reactor pressures. The OPERABILITY of the scram accumulators is based on maintaining adequate accumulator pressure.

BASES

APPLICABILITY

In MODES 1 and 2, the scram function is required for mitigation of DBAs and transients, and therefore the scram accumulators must be OPERABLE to support the scram function. In MODES 3 and 4, control rods are not able ~~only allowed~~ to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram accumulator OPERABILITY during these conditions. Requirements for scram accumulators in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY - Refueling." (1)

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each control rod scram accumulator. This is acceptable since the Required Actions for each Condition provide appropriate compensatory actions for each ~~affected~~ inoperable accumulator. Complying with the Required Actions may allow for continued operation and subsequent ~~affected~~ inoperable accumulators governed by subsequent Condition entry and application of associated Required Actions. (1) (1) (1)

A.1 and A.2

With one control rod scram accumulator inoperable and the reactor steam dome pressure ≥ 900 psig, the control rod may be declared "slow," since the control rod will still scram at the reactor operating pressure but may not satisfy the required scram times in Table 3.1.4-1. Required Action A.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance ~~test~~. Otherwise, the control rod may ~~would~~ already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action A.2) and LCO 3.1.3 is entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function, in accordance with ACTIONS of LCO 3.1.3. (1) (2)

The allowed Completion Time of 8 hours is reasonable, based on the large number of control rods available to provide the scram function and the ability of the affected control rod to scram only with reactor pressure at high reactor pressures.

BASES

ACTIONS (continued)

B.1, B.2.1, and B.2.2

With two or more control rod scram accumulators inoperable and reactor steam dome pressure ≥ 900 psig, adequate pressure must be supplied to the charging water header. With inadequate charging water pressure, all of the accumulators could become inoperable, resulting in a potentially severe degradation of the scram performance. Therefore, within 20 minutes from discovery of charging water header pressure < 940 psig concurrent with Condition B, adequate charging water header pressure must be restored. The allowed Completion Time of 20 minutes is reasonable, to place a CRD pump into service to restore the charging header pressure, if required. This Completion Time is based on the ability of the reactor pressure alone to fully insert all control rods.

The control rod may be declared "slow," since the control rod will still scram using only reactor pressure, but may not satisfy the times in Table 3.1.4-1. Required Action B.2.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control rod scram time is within the limits of Table 3.1.4-1 during the last scram time Surveillance ~~test~~. Otherwise, the control rod may ~~would~~ already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action B.2.2) and the ACTIONS of LCO 3.1.3 entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function in accordance with ACTIONS of LCO 3.1.3. 2 1 1

The allowed Completion Time of 1 hour is reasonable, based on the ability of only the reactor pressure to scram the control rods and the low probability of a DBA or transient occurring while the affected accumulators are inoperable.

C.1 and C.2

With one or more control rod scram accumulators inoperable and the reactor steam dome pressure < 900 psig, the pressure supplied to the charging water header must be adequate to ensure that accumulators remain charged. With the reactor steam dome pressure < 900 psig, the function of the accumulators in providing the scram force becomes much more important since the scram function could become severely

BASES

ACTIONS (continued)

degraded during a depressurization event or at low reactor pressures. Therefore, immediately upon discovery of charging water header pressure < 940 psig, concurrent with Condition C, all control rods associated with inoperable accumulators must be verified to be fully inserted. Withdrawn control rods with inoperable accumulators may fail to scram under these low pressure conditions. The associated control rods must also be declared inoperable within 1 hour. The allowed Completion Time of 1 hour is reasonable for Required Action C.2, considering the low probability of a DBA or transient occurring during the time that the accumulator is inoperable.

D.1

The reactor mode switch must be immediately placed in the shutdown position if either Required Action and associated Completion Time associated with loss of the CRD ~~charging~~ pump (Required Actions B.1 and C.1) cannot be met. This ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. This Required Action is modified by a Note stating that the action is not applicable if all control rods associated with the inoperable scram accumulators are fully inserted, since the function of the control rods has been performed. 4

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

SR 3.1.5.1 requires that the accumulator pressure be checked every 7 days to ensure adequate accumulator pressure exists to provide sufficient scram force. The primary indicator of accumulator OPERABILITY is the accumulator pressure. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1100 psig ~~(Ref. 1)~~. Declaring the accumulator inoperable when the minimum pressure is not maintained ensures that significant degradation in scram times does not occur. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room. 4

REFERENCES 4 U

1. U~~F~~SAR, ~~Section [4.2.3.2.2.4]~~ Chapter 3. 3
2. U~~F~~SAR, ~~Section [5A.4.3]~~ Chapter 14.
3. U~~F~~SAR, ~~Section [15.1]~~ Chapter 14A.

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.5 BASES, CONTROL ROD SCRAM ACCUMULATORS

1. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

2. Typographical/grammatical error corrected.

3. The brackets are removed and the proper plant specific information/value is provided.

4. Changes are made (additions, deletions, and/or changes) to the ISTS Bases, which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.5, CONTROL ROD SCRAM ACCUMULATORS

There are no specific NSHC discussions for this Specification.

ATTACHMENT 6

ITS 3.1.6, Rod Pattern Control

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	(b) when the rod is withdrawn the first time subsequent to each refueling outage, observe discernible response of the nuclear instrumentation. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is critical shall be performed to observe nuclear instrumentation response.
2. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all operable control rods are fully inserted and Specification 3.3.A.1 is met.	2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.
LCO 3.1.6 3.(a) ~~Control rod withdrawal sequences shall be established so that the maximum calculated reactivity that could be added by dropout of any increment of any one control blade will not make the core more than 1.3% Δk supercritical.~~	3.(a) To consider the rod worth minimizer operable, the following steps must be performed: (i) The control rod withdrawal sequence for the rod worth minimizer computer shall be verified as correct. (ii) The rod worth minimizer computer on-line diagnostic test shall be successfully completed. (iii) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

See ITS 3.1.3

See CTS 3/4.3.B.2

See ITS 3.3.2.1

OPERABLE control rods shall comply with the requirements of the banked position withdrawal sequence (BPWS)

A.2

Add proposed Applicability

Add proposed SR 3.1.6.1

M.1

LA.1

L.1

A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
F. Scram Discharge Volume	F. Scram Discharge Volume
1. During reactor operation, the scram discharge volume vent and drain valves shall be operable, except as specified below.	The scram discharge volume vent and drain valves shall be cycled quarterly.
2. If any scam discharge volume vent or drain valve is made or found inoperable, the integrity of the scram discharge volume shall be maintained by either:	Once per operating cycle verify the scram discharge volume vent and drain valves close within 30 seconds after receipt of a reactor scram signal and open when the scram is reset.
a. Verifying daily, for a period not to exceed 7 days, the operability of the redundant valve(s), or	
b. Maintaining the inoperable valve(s), or the associated redundant valve(s), in the closed position. Periodically the inoperable and the redundant valve(s) may both be in the open position to allow draining the scram discharge volume.	
If a or b above cannot be met, at least all but one operable control rods (not including rods removed per specification 3.10.E or inoperable rods allowed by 3.3.A.2) shall be fully inserted within ten hours.	

See ITS 3.1.8

A.3

ACTIONS A and B

G. Required Action

1. If Specifications ~~3.3.A through D above are not met, an orderly shutdown shall be initiated and~~ have reactor in the cold ~~shutdown~~ condition within 24 hours.

(except when the reactor mode switch is in the Refuel position) A.3

Add proposed ACTIONS A and B L.2

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DISCUSSION OF CHANGES
ITS 3.1.6, ROD PATTERN CONTROL

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.3.B.3.(a) states "Control rod withdrawal sequences shall be established so that the maximum calculated reactivity that could be added by dropout of any increment of any one control blade will not make the core more than 1.3% Δk supercritical." Implicit in this requirement is that once the control rod withdrawal sequence is established it will be maintained. ITS LCO 3.1.6 states "OPERABLE control rods shall comply with the requirements of the banked position withdrawal sequence (BPWS)." This changes the CTS by requiring a control rod withdrawal sequence to be continuously met by clarifying the actual control rod withdrawal sequence being used at Monticello. The change that relocates the details of the system design of control rod withdrawal sequences is discussed in DOC LA.1.

The purpose of ITS LCO 3.1.6 is to provide the explicit requirements of the actual required control rod withdrawal sequence that must be used at Monticello. The change is acceptable because the Monticello USAR currently assumes the unit is utilizing the BPWS in the control rod drop accident analysis. This change is designated as administrative because it does not result in any technical changes to the CTS.

A.3 These changes to CTS 3.3.G are provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the USNRC for approval in NMC letter L-MT-05-013, from Thomas J. Palmisano (NMC) to USNRC, dated April 12, 2005. As such, these changes are administrative.

MORE RESTRICTIVE CHANGES

M.1 CTS 4.3.B.3.(a) does not require any verification of proper control rod sequence. ITS SR 3.1.6.1 requires verification that all OPERABLE control rods comply with bank position withdrawal sequence (BPWS) every 24 hours. This changes the CTS by adding a Surveillance requirement to verify all OPERABLE control rods comply with BPWS.

This change is acceptable because it requires a verification to ensure all OPERABLE control rods comply with BPWS. This verification gives additional confidence that the control rod withdrawal sequence is within the bounds assumed in the control rod drop accident. This change is designated as more restrictive because it adds a Surveillance Requirement that is not required in the CTS.

DISCUSSION OF CHANGES
ITS 3.1.6, ROD PATTERN CONTROL

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.3.B.3.(a) states "Control rod withdrawal sequences shall be established so that the maximum calculated reactivity that could be added by dropout of any increment of any one control blade will not make the core more than 1.3% Δk supercritical." ITS LCO 3.1.6 states "OPERABLE control rods shall comply with the requirements of the banked position withdrawal sequence (BPWS)." This changes the CTS by relocating the details of the system design of control rod withdrawal sequences to the USAR.

The removal of this detail, which is related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains that OPERABLE control rods be in compliance with the BPWS. Compliance with the BPWS will ensure the maximum reactivity limit of 1.3% Δk is met. Also, this change is acceptable because the removed information will be adequately controlled in the USAR. The USAR is controlled under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 2 – Relaxation of Applicability)* CTS 3.3.B.3.(a) requires the control rod withdrawal sequences to be established but does not explicitly specify the Applicability of the control rod withdrawal sequences. However, CTS 3.3.G.1 requires the unit to be in cold shutdown (MODE 4) within 24 hours if CTS 3.3.B.3.(a) is not met. Thus this implicitly requires the control rod withdrawal sequence to be met in MODES 1, 2, and 3. ITS LCO 3.1.6 requires all OPERABLE control rods to be in compliance with the bank position withdrawal sequence in MODES 1 and 2 with THERMAL POWER ≤ 10% RTP. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

The purpose of CTS 3.3.B.3.(a) is to ensure the control rod withdrawal sequences are established so that the consequences of a control rod drop accident are within the bounds of the safety analysis. This change is acceptable because the control rod drop accident (CRDA) is relevant at THERMAL POWER ≤ 10% RTP. CTS 3.3.B.3.(a) implies the Applicablity includes MODES 1, 2, and 3 since the default action (CTS 3.3.G.1) requires a shutdown to cold shutdown (MODE 4). At THERMAL POWER > 10% RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel design limit during a CRDA. In MODES 3, 4, and 5, since the reactor is shut

DISCUSSION OF CHANGES
ITS 3.1.6, ROD PATTERN CONTROL

down and, at most, only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable (because the reactor will remain subcritical with a single control rod withdrawn). This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

L.2 *(Category 4 – Relaxation of Required Action)* CTS 3.3.G.1 requires the unit to be in cold shutdown (MODE 4) within 24 hours if the requirement of CTS 3.3.B.3.(a) (control rod withdrawal sequence requirement) is not met. ITS 3.1.6 ACTION A covers the condition when one or more OPERABLE control rods are not in compliance with BPWS, and requires the associated control rod(s) to be moved to the correct position or to declare the associated control rod(s) inoperable within 8 hours. A Note is included for ITS 3.1.6 Required Action A.1 that states the rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation." ITS 3.1.6 ACTION B covers the condition when nine or more OPERABLE control rods are not in compliance with BPWS, and requires the immediate suspension of control rod withdrawal and requires the reactor mode switch to be placed in the shutdown position within 1 hour. A Note similar to the one for ITS 3.1.6 Required Action A.1 is included for ITS 3.1.6 Required Action B.1. This changes the CTS by adding specific ACTIONS for OPERABLE control rods not in compliance with BPWS, in lieu of a shutdown to MODE 4.

The purpose of CTS 3.3.G.1 is to place the unit in a condition in which the LCO does not apply. However, the Applicability of CTS 3.3.B.3.(a) was changed as described in DOC L.1. The purpose of the ITS 3.1.6 ACTIONS is to provide a short period of time to comply with BPWS or to declare the rods inoperable. In addition, the ITS 3.1.6 ACTIONS limit the total number of control rods not in compliance with BPWS, and if this number is exceeded, will also require exiting the Applicability of the LCO. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABILITY status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of an accident occurring during the repair period. The ITS 3.1.6 ACTIONS provide a short period of time to comply with BPWS or to declare the rods inoperable. In addition, the ITS 3.1.6 ACTIONS limit the total number of control rods not in compliance with BPWS. With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, actions may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to ≤ 10% RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight, to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence. ITS 3.1.6 Required Action A.1 is modified by a Note that allows the RWM to be bypassed to allow

DISCUSSION OF CHANGES
ITS 3.1.6, ROD PATTERN CONTROL

the affected control rods to be returned to their correct position. ITS LCO 3.3.2.1 requires verification of control rod movement by a second qualified member of the technical staff. This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by ITS 3.1.6 Required Action A.2. OPERABILITY of control rods is determined by compliance with LCO 3.1.3, "Control Rod OPERABILITY," LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators." The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence. If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. ITS 3.1.6 Required Action B.1 is also modified by a Note similar to the Note for ITS 3.1.6 Required Action A.1 and is acceptable for the same reasons. When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Rod Pattern Control

3.3.B.3.(a) LCO 3.1.6 OPERABLE control rods shall comply with the requirements of the [banked position withdrawal sequence (BPWS)]. 1

DOC L.1 APPLICABILITY: MODES 1 and 2 with THERMAL POWER ≤ [10]% RTP. 1

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.3.G.1	A. One or more OPERABLE control rods not in compliance with [BPWS].	A.1 -------------NOTE------------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation." ---------------------------------		1
		Move associated control rod(s) to correct position.	8 hours	
		OR		
		A.2 Declare associated control rod(s) inoperable.	8 hours	
3.3.G.1	B. Nine or more OPERABLE control rods not in compliance with [BPWS].	B.1 -------------NOTE------------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1. ---------------------------------		1
		Suspend withdrawal of control rods.	Immediately	
		AND		

CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.3.G.1		B.2 Place the reactor mode switch in the shutdown position.	1 hour

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
DOC M.1	SR 3.1.6.1	Verify all OPERABLE control rods comply with [BPWS].	24 hours	1

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.6, ROD PATTERN CONTROL

1. The brackets are removed and the proper plant specific information/value is provided.

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Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to [10]% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA). (1)

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1, ~~and~~ 2, and 3. (2)

APPLICABLE SAFETY ANALYSES

The analytical methods and ~~assumptions~~ used in evaluating the CRDA are summarized in References 1, ~~and~~ 2, and 3. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated. (2)

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. (2) Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. ~~3~~ 4), the fuel design ~~damage~~ limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Refs. ~~4 and~~ 5). Generic evaluations (Refs. ~~1 and 6~~ 6 and 7) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. ~~7~~ 8) and the calculated offsite doses will be well within the required limits (Ref. ~~5~~ 9). (2)

Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to [10]% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. (1)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Generic ~~analysis~~ design of the BPWS (Ref. 1) has demonstrated that the 280 cal/gm fuel ~~damage~~ limit will not be violated during a CRDA while following the BPWS ~~MODE~~ of operation. The generic BPWS analysis (Ref. ~~8~~ 10) also evaluates the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods. 2 7 2

Rod pattern control satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.

APPLICABILITY

In MODES 1 and 2, when THERMAL POWER is ≤ [10]% RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is > [10]% RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel ~~damage~~ design limit during a CRDA (Ref. 2). INSERT 1 In MODE~~S 3, 4, and~~ 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn. 1 1 2 3

ACTIONS

A.1 and A.2

With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, actions may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to ≤ [10]% RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight, to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence. ~~When the control rod pattern is not in compliance with the prescribed sequence, all control rod movement should be stopped except for moves needed to correct the rod pattern, or scram if warranted.~~ 1 4

3

INSERT 1

In MODES 3 and 4, the reactor is shut down and the control rods are not able to be withdrawn since the reactor mode switch is in the shutdown position and a control rod block is applied, therefore a CRDA is not postulated to occur.

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BASES

ACTIONS (continued)

INSERT 2 5

3

Required Action A.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a qualified member of the technical staff. This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. OPERABILITY of control rods is determined by compliance with LCO 3.1.3, "Control Rod OPERABILITY," LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators." The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

helps to

INSERT 3

3 3 6

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a qualified member of the technical staff.

5

INSERT 2

INSERT 3

3

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ [10]% RTP. 1

5

INSERT 2

second licensed operator (Operator or Senior Operator) or by a

3

INSERT 3

(e.g., engineer)

BASES

REFERENCES

1. NEDE-24011-P-A~~-9-US~~, "General Electric Standard Application for Reactor Fuel~~, Supplement for United States," Section 2.2.3.1, September 1988~~. (revision specified in Specification 5.6.3) INSERT 4 2

2. ~~"Modifications to the Requirements for Control Rod Drop Accident Mitigating System," BWR Owners Group, July 1986.~~ 2

3. USAR, Section 14.7.1. 2

2 4 ~~3~~. NUREG-0979, Section 4.2.1.3.2, April 1983.

2 5 ~~4~~. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

2 9 ~~5~~. 10 CFR 100.11.

~~6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.~~

INSERT 5 2

2 8 ~~7~~. ASME, Boiler and Pressure Vessel Code.

2 10 ~~8~~. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

2

INSERT 4

Letter from T.A. Pickens (BWROG) to G.C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.

2

INSERT 5

7. NEDO-10527, "Rod Drop Accident Analysis for Large BWRs," (including Supplements 1 and 2), March 1972.

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.6 BASES, ROD PATTERN CONTROL

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

4. This requirement has been deleted since ACTION A does not require that all rod movement (except for the moves needed to correct the rod pattern or a scram) be suspended.

5. Changes have been made to more clearly match the requirements of ITS 3.3.2.1 Required Action C.2.2.

6. A reference to the location where control rod OPERABILITY is determined has been deleted from the Bases for Required Actions A.1 and A.2 of ITS 3.1.6. This section is discussing under what conditions related to control rod sequence to declare a control rod inoperable - not determination of OPERABILITY per the other LCOs. As such, the reference is not applicable and could be interpreted as requiring an action that is not in the actual ITS 3.1.6 ACTION A.

7. Typographical error corrected.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.6, ROD PATTERN CONTROL

There are no specific NSHC discussions for this Specification.

ATTACHMENT 7

ITS 3.1.7, Standby Liquid Control (SLC) System

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS | ITS

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3.0 LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

~~Applicability:~~

~~Applies to the operating status of the standby liquid control system.~~

~~Objective:~~

~~To assure the availability of an independent reactivity control mechanism.~~

Specification:

A. System Operation (A.2)

LCO 3.1.7 — 1. The standby liquid control system shall be operable ~~at all times when fuel is in the reactor and the reactor is not shut down by control rods,~~ [Applicability: MODES 1 and 2] except as specified in 3.4.A.2.

ACTION B — 2. From and after the date that a redundant component is made or found to be inoperable, reactor operation is permissible only during the following 7 days provided that the redundant component is operable.

Add proposed ACTION C (L.1)

4.0 SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

~~Applicability:~~

~~Applies to the periodic testing requirements for the standby liquid control system.~~

~~Objective:~~

~~To verify the operability of the standby liquid control system.~~

Specification:

A. The operability of the standby liquid control system shall be verified by performance of the following tests:

1. ~~At least once per quarter -~~ (A.3)

SR 3.1.7.7 — Pump minimum flow rate of 24 gpm shall be verified against a system head of 1275 psig when tested in accordance with the Inservice Testing Program. ~~Comparison of the measured pump flow rate against equation 2 of paragraph 3.4.B.1 shall be made to demonstrate operability of the system in accordance with the ATWS Design Basis.~~ (L.2)

2. At least once during ~~each operating cycle~~ [24 months] (A.5)

SR 3.1.7.8 — a. ~~Manually initiate~~ [Verify flow through] one of the two standby liquid control systems ~~and pump demineralized water~~ [from pump] into the reactor vessel. ~~This test checks explosion of the charge associated with the tested system, proper operation of the valves and pump capacity.~~ (LA.1) Both systems shall be tested ~~and inspected, including each explosion valve~~ (A.6) ~~in the course of two operating cycles~~ [on a STAGGERED TEST BASIS] (A.1)

Add proposed ITS SR 3.1.7.4 and SR 3.1.7.6, SR 3.1.7.9 (M.2)

A.1

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	b. ~~Explode one of two primer assemblies manufactured in the same batch to verify proper function. Then install, as a replacement, the second primer assembly in the explosion valve of the system tested for operation.~~

LA.1

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A.1

ITS

ITS

3.0 LIMITING CONDITIONS FOR OPERATION

LCO 3.1.7 — B. ~~Boron Solution Requirements~~

~~At all times when the Standby Liquid Control System is required to be operable:~~

SR 3.1.7.1, SR 3.1.7.5, SR 3.1.7.10 — 1. The liquid poison tank shall contain a boron bearing solution that satisfies the volume, concentration and enrichment requirements of Figure 3.4.1, or compliance can be demonstrated by satisfying the following equations:

Equation 1 (Original Design Basis):

$$V \ge \left(\frac{71.18}{0.0051 \times C + 0.998}\right)\left(1+\frac{4821}{1101-E}\right)\left(\frac{19.8}{E}\right)\left(\frac{100}{C}\right) + 128 \text{ gal}$$

Equation 2 (ATWS Design Basis):

LA.2

SR 3.1.7.1, SR 3.1.7.5, Table 3.1.7-1

$$C \ge 8.28 \left(\frac{86}{Q}\right)\left(\frac{19.8}{E}\right)$$

where:

V - indicated Boron solution tank volume (gal)
E - measured Boron solution enrichment (atom%)
C - measured Boron solution concentration (wt%)
Q - measured pump flow rate (gpm) at 1275 psig

ACTION A — If Equation 1 is satisfied but Equation 2 cannot be met, continued plant operation is permissible, provided that:

a. Compliance with Equation 2 is demonstrated within 7 days ~~or~~

b. ~~The Commission shall be notified and a special report provided outlining the actions taken and the plans and schedule for demonstrating compliance with the ATWS Design Basis.~~

M.3

SR 3.1.7.2 — 2. The temperature shall not be less than the solution temperature presented in Figure 3.4.2.

SR 3.1.7.3 — 3. The heat tracing on the pump suction lines shall be operable whenever the room temperature is less than the solution temperature presented in Figure 3.4.2.

4.0 SURVEILLANCE REQUIREMENTS

B. Boron Solution Surveillance

~~The availability of the proper boron bearing solution shall be verified by performance of the following tests:~~ L.3

1. ~~At least once per cycle~~ - A.4

SR 3.1.7.10 — Boron enrichment shall be determined. ≥ 55.0 atom percent B-10 ~~The laboratory analysis to determine enrichment shall be obtained within 30 days of sampling or chemical addition.~~ Prior to addition to SLC tank L.3

LA.3

2. At least once per month -

SR 3.1.7.5 — Boron concentration shall be determined. In addition, the boron concentration shall be determined ~~any time~~ Once within 24 hours after water or boron are added ~~or if~~ and once within 24 hours after the solution temperature ~~drops below~~ is restored within the limits specified by Figure 3.4.2. M.4

3. At least once per day -

SR 3.1.7.1 — a. Solution volume shall be checked.

SR 3.1.7.2 — b. The solution temperature shall be checked.

SR 3.1.7.3 — c. The room temperature shall be checked in the vicinity of the standby liquid control system pumps.

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ITS

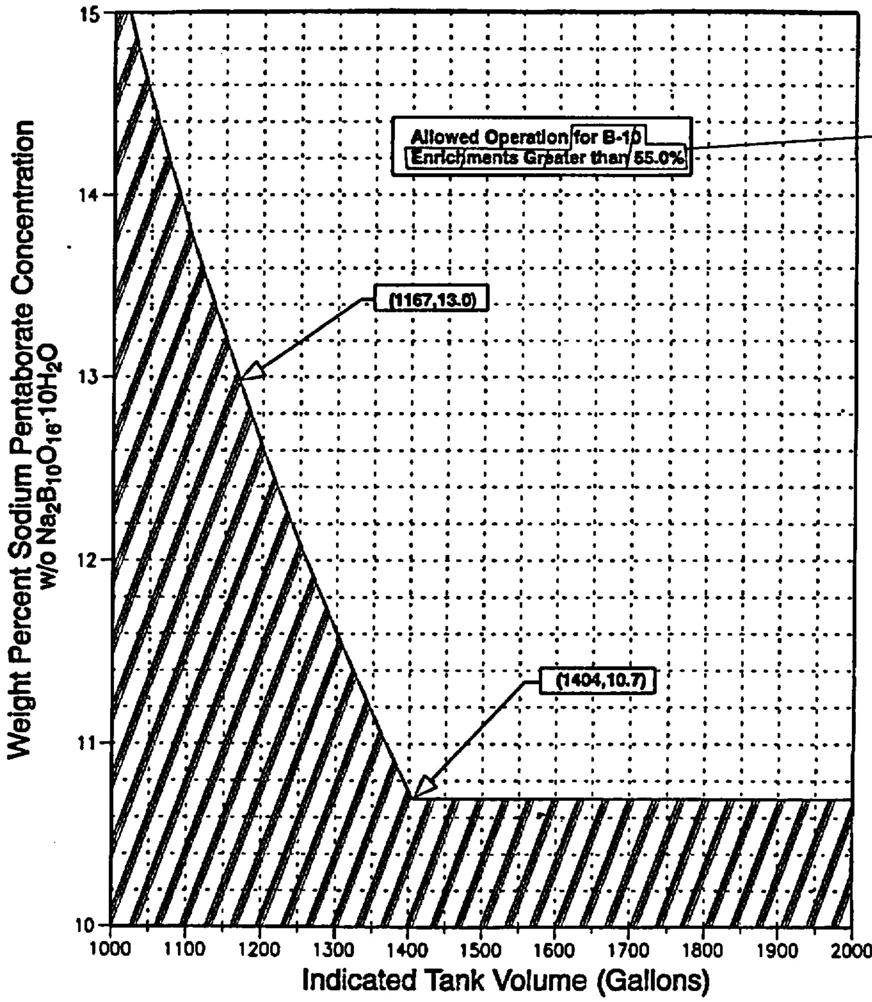
3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
ACTION D — C. If Specification 3.4.A through B are not met, an orderly shutdown shall be initiated and the reactor shall be in Hot Shutdown within 12 hours.	

ITS 3.1.7

A.1

ITS

Figure 3.1.7-1



LA.2

Amendment No. 57

Figure 3.4-1 Sodium Pentaborate Solution Volume Concentration Requirements

97 9/23/88
Amendment No. 57

A.1

ITS

Figure 3.1.7-2

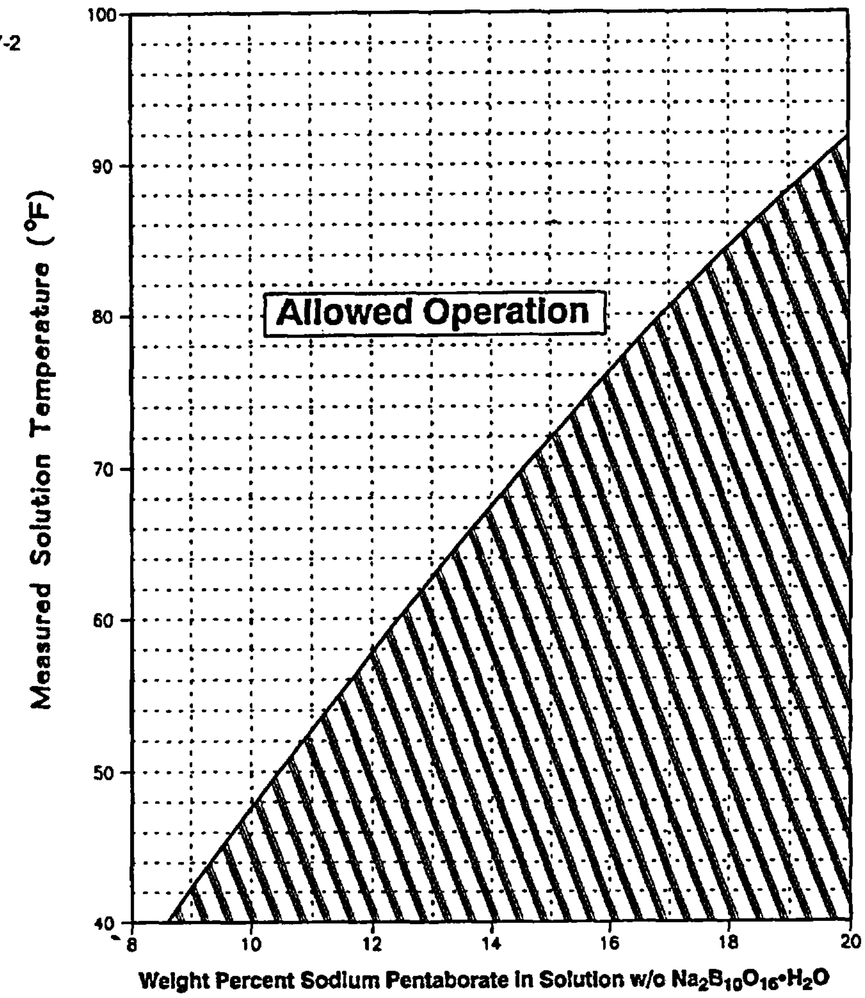


Figure 3.4-2 Sodium Pentaborate Solution Temperature Requirements

98 12/11/87
Amendment No. 56

DISCUSSION OF CHANGES
ITS 3.1.7, STANDBY LIQUID CONTROL (SLC) SYSTEM

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.4.A.1 requires the Standby Liquid Control (SLC) System to be OPERABLE at all times when fuel is in the reactor and the reactor is not shut down by control rods. ITS LCO 3.1.7 requires the SLC System to be OPERABLE in MODES 1 and 2. This changes the CTS by explicitly stating the applicable MODES in which the SLC System must be OPERABLE.

The purpose of the CTS 3.4.A.1 is to ensure the SLC System is available to shutdown the reactor core whenever it is not shut down (i.e., multiple control rods are withdrawn). ITS 3.1.7 only requires the SLC System to be OPERABLE in MODES 1 and 2. This change is acceptable because MODES 1 and 2 are the only MODES in which the reactor is not shut down by control rods. In MODES 3 and 4, control rods cannot be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies, and LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," ensures that the reactor will remain in shutdown by use of control rods. This clarification is also consistent with CTS 3.4.C, which requires a unit shutdown to Hot Shutdown (MODE 3) if CTS 3.4.A or B is not met. This change is designated as administrative since it does not result in any technical changes to the CTS.

A.3 CTS 4.4.A.1 specifies those Surveillance Requirements that must be performed "At least once per quarter." CTS 4.4.A.1 only requires the performance of a SLC System flow test. However, CTS 4.4.A.1 also states that the SLC System flow test must be performed "in accordance with the Inservice Testing Program." ITS SR 3.1.7.7 requires the same test to be performed "In accordance with the Inservice Testing Program." This changes the CTS by deleting the duplicative information associated with the testing Frequency.

The purpose of the CTS 4.4.A.1 is to perform the SLC System flow test in accordance with the Inservice Testing Program. This changes the CTS by deleting the duplicative information associated with the testing Frequency. This change is acceptable since currently the Frequency of pump tests in the Inservice Testing Program is 92 days. This change simply deletes duplicative testing Frequencies from the CTS. Therefore, this change is considered a presentation preference change only and, as such, is considered an administrative change.

A.4 CTS 4.4.B.1 requires a determination of boron enrichment, but does not specify the actual limit. The design limit for Monticello is 55.0 atom percent, as stated in CTS Figure 3.4-1. ITS SR 3.1.7.10 requires verification that the sodium

DISCUSSION OF CHANGES
ITS 3.1.7, STANDBY LIQUID CONTROL (SLC) SYSTEM

pentaborate enrichment is ≥ 55.0 atom percent. This changes the CTS by specifying the actual limit in the sodium pentaborate enrichment verification Surveillance.

The purpose of CTS 4.4.B.1 is to verify the sodium pentaborate enrichment is within the design limit so that Figure 3.4-1, which is based on a boron enrichment of 55.0 atom percent, can be used. Therefore, this change is acceptable since the limit specified in CTS Figure 3.4-1 is being added to the appropriate Surveillance. This change is considered a presentation preference change only and, as such, is considered an administrative change.

A.5 CTS 4.4.A.2 requires the performance of a SLC System test at least once "during each operating cycle." ITS SR 3.1.7.8 requires performance of an SLC test every "24 months" on a STAGGERED TEST BASIS. This changes the CTS by changing the Frequency from "during each operating cycle" to "24 months."

This change is acceptable because the current "operating cycle" is "24 months." In letter L-MT-04-036, from Thomas J. Palmisano (NMC) to the USNRC, dated June 30, 2004, NMC has proposed to extend the fuel cycle from 18 to 24 months and the same time has performed an evaluation in accordance with Generic Letter 91-04 to extend the unit Surveillance Requirements from 18 months to 24 months. CTS 4.4.A.2 was included in this evaluation. This change is designated as administrative because it does not result in any technical changes to the CTS.

A.6 CTS 4.4.A.2.a requires the performance of a SLC subsystem test to verify flow can be injected into the reactor vessel. During this test SLC pump capacity must be verified. ITS SR 3.1.7.8 requires the performance of the same test, however the requirement to verify pump capacity has not been included. This changes the CTS by deleting the specific requirement to verify SLC pump capacity during the SLC subsystem reactor vessel injection test.

The purpose of CTS 4.4.A.2.a is to ensure the flow path from the pump to the reactor vessel is not obstructed. This change deletes the specific requirement to verify pump capacity during the SLC subsystem reactor vessel injection test. This change is acceptable because SLC pump capacity is verified more frequently as required by CTS 4.4.A.1, the quarterly pump capacity test. This Surveillance is maintained in the ITS as SR 3.1.7.7 at a Frequency in accordance with the Inservice Testing Program (currently every 92 days). This test will ensure that the SLC pump capacity is adequate to perform its safety function. ITS SR 3.1.7.8 requires the verification of flow through one SLC subsystem from pump into the reactor pressure vessel, and is sufficient to ensure the piping from the pump to the reactor vessel is not obstructed. The requirement to verify SLC pump capacity is duplicative and is therefore deleted from CTS 4.4.A.2.a. As such, this change is considered a presentation preference change only and is therefore designated as an administrative change.

DISCUSSION OF CHANGES
ITS 3.1.7, STANDBY LIQUID CONTROL (SLC) SYSTEM

MORE RESTRICTIVE CHANGES

M.1 Not used.

M.2 ITS SR 3.1.7.4 requires the verification of the continuity of the explosive charge. ITS SR 3.1.7.6 requires verification that each SLC subsystem manual valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position. ITS SR 3.1.7.9 requires verification that all heat traced piping between storage tank and pump suction is unblocked. The CTS does not include these Surveillance Requirements. This changes the CTS by adding these new Surveillances.

This change is acceptable because the new Surveillance Requirements will help ensure the SLC System is OPERABLE. These verifications give additional confidence that the explosive valves are OPERABLE, the SLC manual valves are aligned correctly (or can be aligned), and that the heat traced piping between the storage tank and pump suction is unblocked. This change is designated as more restrictive because it adds Surveillance Requirements that are not required in the CTS.

M.3 With boron concentration limits of CTS 3.4.B.1 not met, CTS 3.4.B.1.a requires compliance with Equation 2 to be demonstrated within 7 days. If compliance with Equation 2 is not demonstrated within 7 days, CTS 3.4.B.1.b requires the Commission to be notified and a special report provided outlining the actions taken and the plans and schedule for demonstrating compliance with the ATWS Design Basis. ITS 3.1.7 ACTION A maintains 7 days to establish the appropriate conditions to satisfy the ATWS Design Basis, but if Equation 2 is not satisfied within the 7 day period, ITS 3.1.7 ACTION D requires a shutdown to MODE 3 within 12 hours. This changes the CTS by deleting the option to notify the Commission and continuing to operate with Equation 2 not met.

The purpose of CTS 3.4.B.1.b is to provide an outline of the actions and schedule to comply with the ATWS Design Basis. The 7 day Completion Time in CTS 3.4.B.1.a (ITS 3.1.7 Required Action A.1) is considered an acceptable amount of time to restore all normal problems associated with the boron solution. If the 7 day Completion Time is not satisfied, ITS 3.1.7 ACTION D will require the unit to be in MODE 3 in 12 hours. This is the current shutdown action in CTS 3.4.C. The change has been designated as more restrictive because it will require the unit to be in MODE 3 in 12 hours instead of allowing operations to continue indefinitely with the ATWS Design Basis not met.

M.4 CTS 4.4.B.2 requires the boron concentration to be determined anytime water or boron is added to the solution or if the solution temperature drops below the limits specified in Figure 3.4-2. However, no finite time to complete performance of this Surveillance is provided. ITS SR 3.1.7.5 requires the same Surveillance; however, a requirement has been added to require the Surveillance to be completed once "within 24 hours" after water or boron is added to the solution and once "within 24 hours after solution temperature is restored" within the limits of Figure 3.1.7-2. This changes the CTS by placing a time limit of 24 hours to perform the Surveillance.

DISCUSSION OF CHANGES
ITS 3.1.7, STANDBY LIQUID CONTROL (SLC) SYSTEM

The purpose of CTS 4.4.B.2 is to ensure the boron concentration is within limits. This change places a time limitation for performing the boron concentration verification after adding water, boron, or after the temperature falls below the temperature limit and is subsequently restored. This change is acceptable because the time limit of 24 hours is sufficient to notify the appropriate personnel to take a sample, send the sample to the laboratory, analyze the sample, and evaluate the results of the chemical analysis. This ensures that any potential change to the boron concentration is quickly evaluated. Also, the second Frequency ensures that the boron concentration is verified "after" the temperature is restored to within limits. The change has been designated as more restrictive because it explicitly limits the time to perform the Surveillance.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.A.2.a states "Manually initiate" one of the two standby liquid control systems "and pump demineralized water" into the reactor vessel. It further states that "This test checks explosion of the charge associated with the tested system, proper operation of the valves" and that both SLC subsystems shall be tested "and inspected, including each explosion valve." CTS 4.4.A.2.b states "Explode one of the primer assemblies manufactured in the same batch to verify proper function. Then install, as a replacement, the second primer assembly in the explosion valve of the system tested for operation." ITS SR 3.1.7.8 requires verification of flow through one SLC subsystem from pump into reactor pressure vessel. This changes the CTS by relocating the above procedural details concerning performance of the flow path test to the ITS Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to verify flow through one SLC subsystem from pump into reactor pressure vessel. Also, this change is acceptable because this type of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because a procedural detail for meeting Technical Specification requirements is being removed from the Technical Specifications.

LA.2 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.4.B.1 states Equation 1 is consistent with the "Original Design Basis" and Equation 2 is consistent with the "ATWS Design Basis." CTS Figure 3.4-1 specifies that the curves are based on "B-10 Enrichments Greater than 55.0%." ITS Figure 3.1.7-1 includes the same requirements as

DISCUSSION OF CHANGES
ITS 3.1.7, STANDBY LIQUID CONTROL (SLC) SYSTEM

CTS Figure 3.4-1, except the detail concerning the B-10 enrichment. ITS Table 3.1.7-1 includes Equation 1 and Equation 2, however the statements concerning the "Original Design Basis" and "ATWS Design Basis" are not included. This changes the CTS by relocating these details to the ITS Bases.

The removal of these details, which are related to the system design capabilities, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the same Figure and equations that are in the CTS. The details on the design basis details of the equations do not need to appear in the specification in order for the requirement to apply. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA.3 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.4.B.1 requires the boron enrichment to be determined by "laboratory analysis." ITS SR 3.1.7.10 does not specify the method that shall be used to determine the B-10 enrichment. This changes the CTS by relocating the procedure detail "laboratory analysis" to the ITS Bases.

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to verify boron enrichment is within limits. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 3 – Relaxation of Completion Time)* CTS 3.4.A does not provide actions for when two SLC subsystems are inoperable, thus CTS 3.4.C must be entered and the unit must be placed in hot shutdown. ITS 3.1.7 ACTION C covers the condition when two SLC subsystems are inoperable for reasons other than Condition A (i.e., boron concentration not within limits), and requires the restoration of one SLC subsystem to OPERABLE status within 8 hours. This changes the CTS by providing 8 hours to restore one SLC subsystem to OPERABLE status when it is discovered that both SLC subsystems are inoperable prior to requiring a unit shutdown.

DISCUSSION OF CHANGES
ITS 3.1.7, STANDBY LIQUID CONTROL (SLC) SYSTEM

The purpose of ITS 3.1.7 ACTION C is to allow 8 hours to restore one SLC subsystem to OPERABLE status when both are inoperable. This change is acceptable because the Completion Time is consistent with safe operation under the specified condition, considering the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a design basis accident occurring during the allowed Completion Time. The ITS 3.1.7 Completion Time of 8 hours is considered acceptable given the low probability of a design basis accident or transient occurring concurrent with the failure of the control rods to shut down the reactor. This change is designated as less restrictive because additional time is allowed to restore a SLC subsystem to OPERABLE status than was allowed in the CTS.

L.2 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.4.A.1 requires the performance of a SLC pump test. It also states "Comparison of the measured pump flow rate against equation 2 of paragraph 3.4.B.1 shall be made to demonstrate operability of the system in accordance with the ATWS Design Basis." ITS SR 3.1.7.7 requires the SLC pump test, but does not include the requirement about the demonstration of the OPERABILITY of the system in accordance with the ATWS Design Basis. This changes the CTS by deleting the requirement to perform this comparison.

The purpose of CTS 4.4.A.1, in part, is to ensure the ATWS Design Basis is met after a flow test is performed. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a Frequency necessary to give confidence that the equipment can perform its assumed safety function. This change deletes the explicit CTS requirement to demonstrate operability of the system in accordance with the ATWS Design Basis after the performance of the SLC pump test. This change is acceptable since there are other Surveillances that are performed more frequently which confirm the ATWS Design Basis. ITS SR 3.1.7.5 requires the verification of the concentration of boron in solution is within the limits of Figure 3.1.7-1 or within the limits of Equation 2 (ATWS Design Basis) of Table 3.1.7-1 every 31 days, once within 24 hours after water or boron is added to solution, and once within 24 hours after solution temperature is restored within limits of Figure 3.1.7-2. The ATWS Design Basis will always be met if the SLC system flow rate is $\geq$ 24 gpm, the sodium pentaborate concentration is within the limits of ITS Figure 3.1.7-1 (CTS Figure 3.4-1), and B-10 enrichment is $\geq$ 55.0%. If the sodium pentaborate concentration is not within the limits of the Figure or if Boron enrichment is < 55.0%, then it is necessary to determine whether the concentration limits of Equation 2 of Table 3.1.7-1 (ATWS Design Basis) is met, as required by ITS SR 3.1.7.5. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L.3 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS 4.4.B.1 requires the boron enrichment to be determined at least once per cycle. The laboratory analysis to determine enrichment shall be obtained within 30 days of sampling or chemical addition. ITS SR 3.1.7.10 requires the determination of B-10 enrichment is $\geq$ 55.0 atom percent B-10 prior to addition to the SLC tank. This changes the CTS by deleting the requirement to verify the

DISCUSSION OF CHANGES
ITS 3.1.7, STANDBY LIQUID CONTROL (SLC) SYSTEM

storage tank enrichment every cycle and replaces it with a requirement to verify that the solution added to the SLC storage tank is at the proper B-10 enrichment.

The purpose of CTS 4.4.B.1 is to ensure the B-10 enrichment in the boron solution tank is within the appropriate limits. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Isotopic tests on the granular sodium pentaborate in the storage container to verify the actual B-10 enrichment must be performed prior to addition to the SLC tank in order to ensure that the proper B-10 atom percentage is being used. This change is acceptable since no deterioration of the B-10 enrichment level should occur to the B-10 while it is stored in its storage container. In addition, the deletion of the requirement to obtain the test results (i.e., the laboratory analysis) within 30 days of sampling or chemical addition is acceptable since the granular B-10 cannot be added to the SLC storage tank until the results of the analysis are known (i.e., the Frequency of ITS SR 3.1.7.10 requires performance "prior to addition"). This change is designed as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

3.4.A.1, 3.4.B LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME	
	CONDITION	REQUIRED ACTION	COMPLETION TIME	3
3.4.B.1, 3.4.B.1.a	A. [Concentration of ~~boron~~ sodium pentaborate in solution not within limits but > ~~[ ]~~ of Figure 3.1.7-1 and Table 3.1.7-1 Equation 2, available volume of sodium pentaborate solution is within limits of Table 3.1.7-1 Equation 1.	A.1 Restore concentration of ~~boron~~ sodium pentaborate in solution to within limits.	~~72 hours~~ 7 days ~~AND~~ ~~10 days from discovery of failure to meet the LCO]~~	2 1 7
3.4.A.2	B. One SLC subsystem inoperable ~~[~~ for reasons other than Condition A ~~]~~.	B.1 Restore SLC subsystem to OPERABLE status.	7 days ~~AND~~ ~~[10 days from discovery of failure to meet the LCO]~~	1 7
DOC L.1	C. Two SLC subsystems inoperable ~~[~~ for reasons other than Condition A ~~]~~.	C.1 Restore one SLC subsystem to OPERABLE status.	8 hours	1
3.4.C	D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	12 hours	

CTS

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
3.4.B.1, 4.4.B.3.a	SR 3.1.7.1	Verify available volume of sodium pentaborate solution is [within the limits of Figure 3.1.7-1] or ~~≥ [4530] gallons~~. Equation 1 of Table 3.1.7-1	24 hours	1
3.4.B.2, 4.4.B.3.b	SR 3.1.7.2	[Verify temperature of sodium pentaborate solution is within the limits of [Figure 3.1.7-2].	24 hours]	1
3.4.B.3, 4.4.B.3.c	SR 3.1.7.3	solution temperature [Verify temperature of ~~pump suction piping~~ room in the vicinity of the SLC pumps is within the limits of [Figure 3.1.7-2] or verify SLC pump suction lines heat tracing is OPERABLE.	24 hours]	1
DOC M.2	SR 3.1.7.4	Verify continuity of explosive charge.	31 days	
3.4.B.1, 4.4.B.2	SR 3.1.7.5	Verify the concentration of ~~boron~~ sodium pentaborate in solution is [within the limits of Figure 3.1.7-1] or within the limits of Equation 2 of Table 3.1.7-1.	31 days AND Once within 24 hours after water or ~~boron~~ sodium pentaborate is added to solution AND Once within 24 hours after solution temperature is restored within the limits of [Figure 3.1.7-2]	1 3 1
DOC M.2	SR 3.1.7.6	Verify each SLC subsystem manual ~~power operated, [and automatic~~ valve] in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	31 days	4

CTS

SURVEILLANCE REQUIREMENTS (continued)

CTS	SR	SURVEILLANCE	FREQUENCY	
4.4.A.1	SR 3.1.7.7	Verify each pump develops a flow rate ≥ 24 ~~[41.2]~~ gpm at a discharge pressure ≥ 1275 ~~[1190]~~ psig.	[In accordance with the Inservice Testing Program ~~or 92 days]~~	1
4.4.A.2	SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 ~~[18]~~ months on a STAGGERED TEST BASIS	1
DOC M.2	SR 3.1.7.9	[Verify all heat traced piping between storage tank and pump suction is unblocked. NOTE Only required if SLC pump suction lines heat tracing is inoperable.	24 ~~[18]~~ months **AND** Once within 24 hours after room ~~solution~~ temperature in the vicinity of the SLC pumps is restored within the solution temperature limits of [Figure 3.1.7-2]]	1 5 1
4.4.B.1	SR 3.1.7.10	[Verify sodium pentaborate enrichment is ≥ 55.0 ~~[60.0]~~ atom percent B-10.	Prior to addition to SLC tank]	1

INSERT 1

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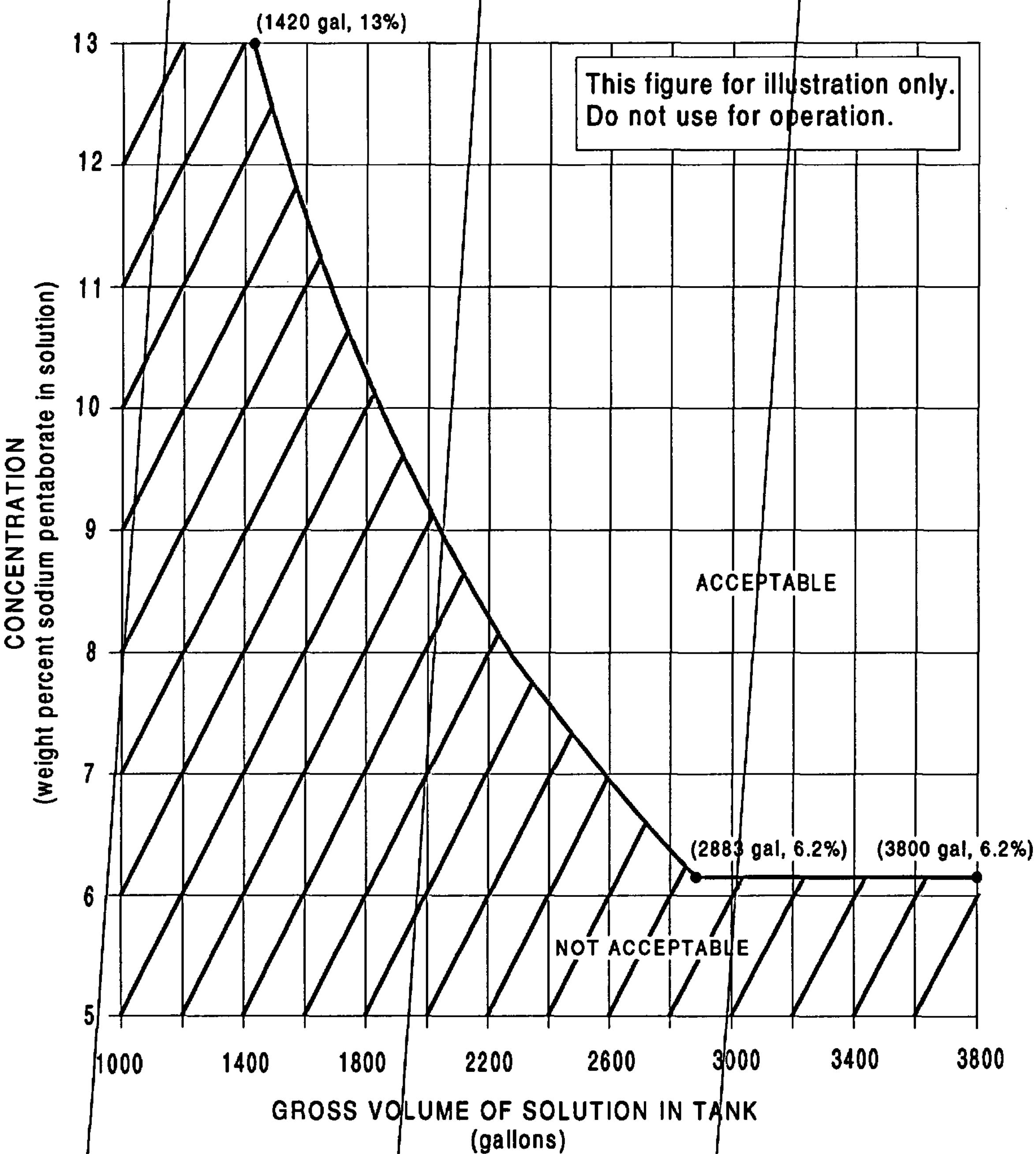

Figure 3.1.7-1 (page 1 of 1)
Sodium Pentaborate Solution Volume
Versus Concentration Requirements

CTS

6

INSERT 1

Figure
3.4-1

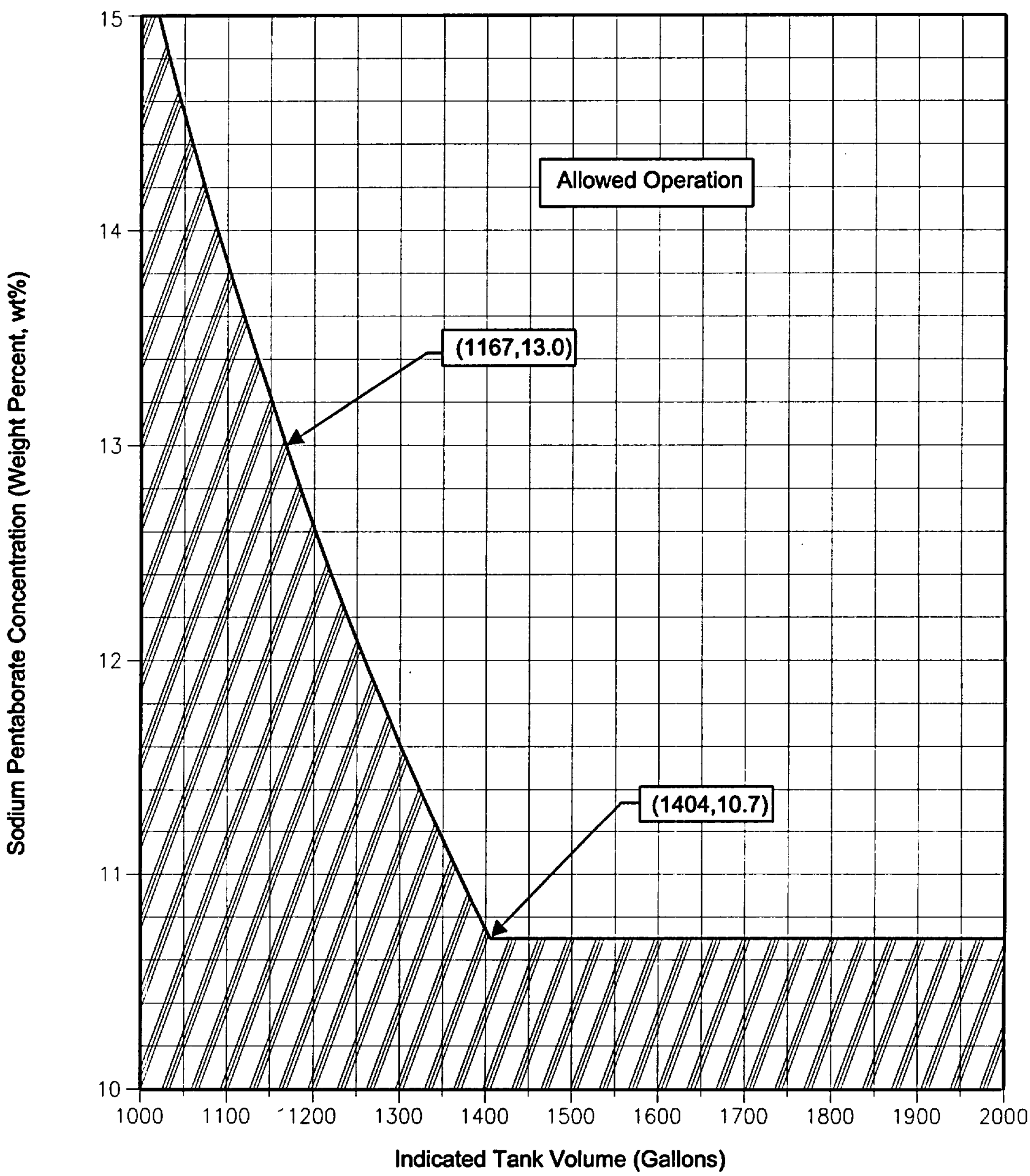

Figure 3.1.7-1 (page 1 of 1)
Sodium Pentaborate Solution Volume Versus Concentration Requirements

SLC System
3.1.7

INSERT 2 6

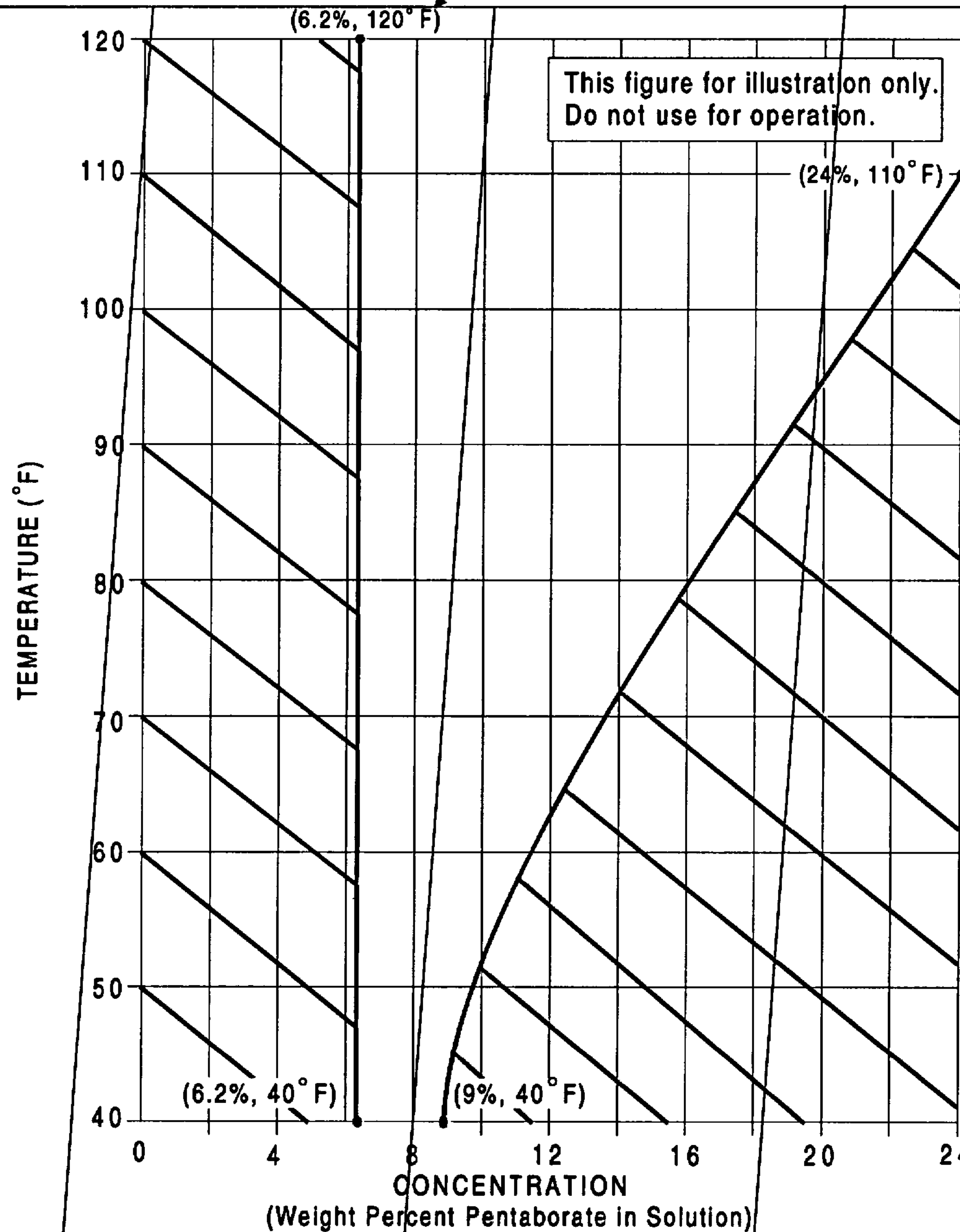

Figure 3.1.7-2 (page 1 of 1)
Sodium Pentaborate Solution Temperature Versus Concentration Requirements

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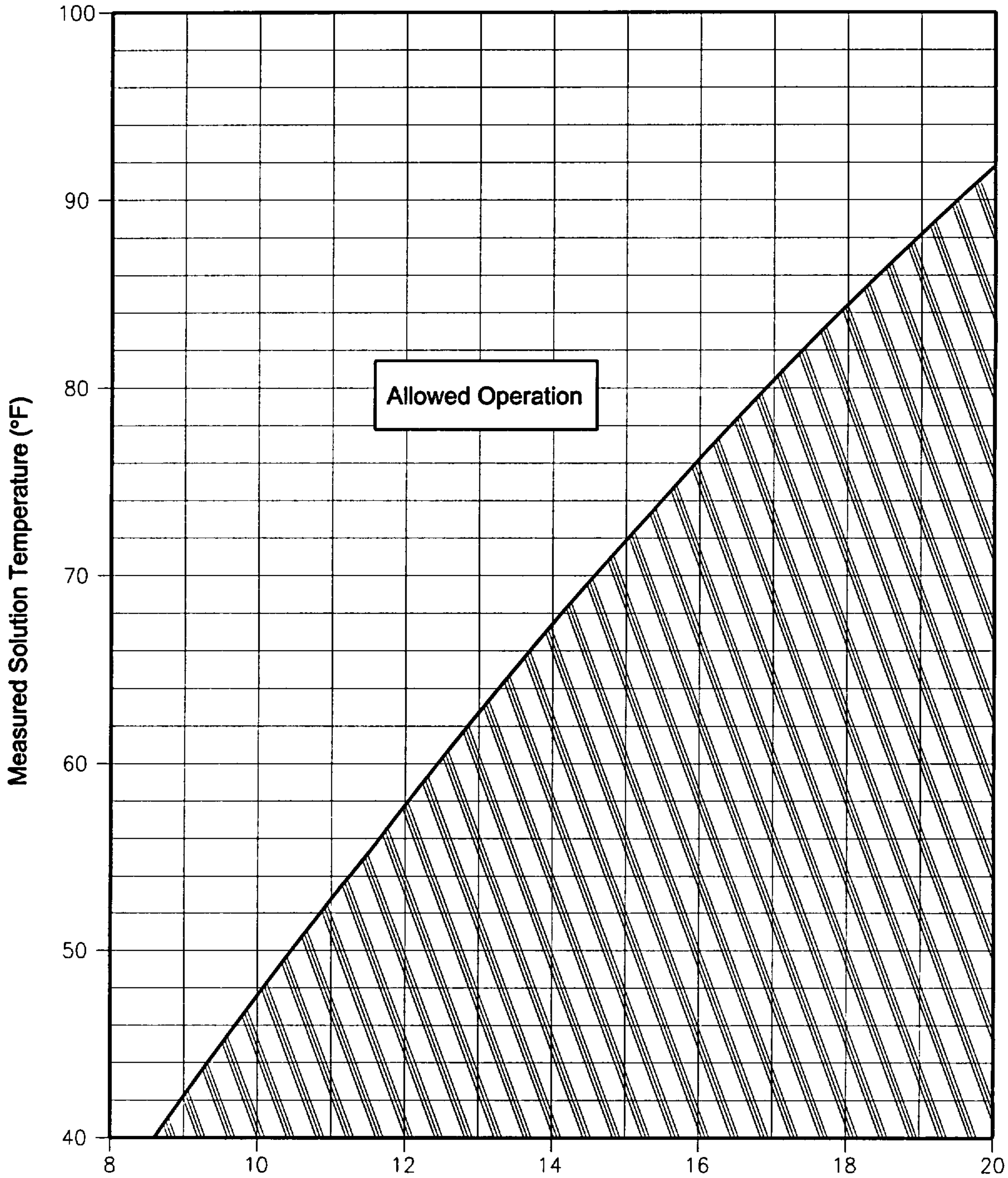
CTS

6 INSERT 2

Figure 3.4-2



Sodium Pentaborate in Solution (Weight Percent, wt%)

Figure 3.1.7-2 (page 1 of 1)
Sodium Pentaborate Solution Temperature Versus Concentration Requirements

3.1.7

6

INSERT 3

CTS

Table 3.1.7-1 (page 1 of 1)
Equations for Required Sodium Pentaborate Tank Volume and Concentration

3.4.B.1 Equation 1

$$V \geq \left(\frac{71.18}{0.0051 \times C + 0.998}\right)\left(1+\frac{4821}{1101-E}\right)\left(\frac{19.8}{E}\right)\left(\frac{100}{C}\right)+128 \text{ gal}$$

Where:

C = measured boron solution concentration (wt%)
E = measured boron solution enrichment (atom%)
V = indicated boron solution tank volume (gal)

Equation 2

$$C \geq 8.28\left(\frac{86}{Q}\right)\left(\frac{19.8}{E}\right)$$

Where:

C = measured boron solution concentration (wt%)
E = measured boron solution enrichment (atom%)
Q = measured pump flow rate (gpm) at 1275 psig

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.7, STANDBY LIQUID CONTROL (SLC) SYSTEM

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. The ISTS 3.1.7 Required Action A.1 first Completion Time has been extended from 72 hours to 7 days, consistent with the current licensing basis (CTS 3.4.B.1.a).

3. The proper Monticello nomenclature has been used (CTS Figures 3.4.-1 and 3.4-2). This is also consistent with the nomenclature used in SR 3.1.7.1 and SR 3.1.7.2.

4. The changes in ITS SR 3.1.7.6 are made since there are no automatic valves in the SLC System and there are no power operated valves other than the explosive valves in the SLC System, and these are not checked as part of this Surveillance (as described in the ISTS Bases for this SR). Explosive valves are tested by other Surveillances in this Specification.

5. ISTS SR 3.1.7.9 requires a verification that all heat traced piping is unblocked once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2. A change in solution temperature in the tank does not necessarily have an impact on the piping temperature, as long as the piping heat trace circuit is functioning properly. The intent of the second Frequency is to ensure that, if the heat tracing is inoperable such that piping temperature falls below the specified minimum temperature, after the heat tracing is restored to OPERABLE status and the piping temperature is greater than or equal to the specified minimum temperature the piping is still unblocked. This is supported by the ISTS SR 3.1.7.9 Bases description for this second Frequency, which describes the requirement as required to be performed after piping temperature is restored. However, since the Monticello design does not include temperature indication on the suction piping, the plant-specific requirement for determining piping temperature, by measuring the room temperature in the vicinity of the SLC pumps, will be used in the Frequency. Thus the second Frequency has been changed to once within 24 hours after "room temperature in the vicinity of the SLC pumps" is restored within the "solution temperature" limits of Figure 3.1.7-2. This plant-specific requirement concerning the room temperature is shown in CTS 3.4.B.3.c and 4.4.B.3.c. Furthermore, a Note has been added stating that the second Frequency is only required if the SLC pump suction lines heat tracing is inoperable, consistent with the above discussed intent.

6. The following changes have been made to reflect the current licensing basis requirements.

7. These changes are made consistent with TSTF-439, Rev. 2, which has been incorporated by the USNRC into Revision 3.1 of NUREG-1433.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND	The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram. (ATWS) 1 The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.
APPLICABLE SAFETY ANALYSES	The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator ~~believes~~ determines 1 the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron ~~which~~ that produces a concentration of 660 ppm of natural boron in the reactor coolant at 68°F. 5 To allow for potential leakage and imperfect mixing in the reactor system, an amount of boron equal to 25% of the amount cited above is added (Ref. 2). The volume versus concentration limits in Figure 3.1.7-1 and the temperature versus concentration limits in Figure 3.1.7-2 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping and with B-10 enrichment of ≥ 55.0 atom percent. This quantity of borated solution is the amount that is above the pump suction ~~shutoff level in the boron solution storage tank~~ nozzle and accounts for wide range instrument accuracy. No credit is taken for the portion of the tank volume that cannot be injected. 2 The SLC System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

ACTIONS

A.1

concentration of sodium pentaborate in

not within

of Figure 3.1.7-1 and Table 3.1.7-1 Equation 2

7 days

(ATWS design basis)

available volume of sodium pentaborate solution is within limits of Table 3.1.7-1 Equation 1

design

2

4

If the ~~boron~~ solution ~~concentration~~ is ~~less than the required~~ limits ~~for mitigation~~ but ~~greater than the concentration required for cold shutdown (original licensing basis)~~, the concentration must be restored to within limits in ~~72 hours~~. It is not necessary under these conditions to enter Condition C for both SLC subsystems inoperable since they are capable of performing their original design basis function. Because of the low probability of an event and the fact that the SLC System capability still exists for vessel injection under these conditions, the allowed Completion Time of ~~72 hours~~ is acceptable and provides adequate time to restore concentration to within limits.

7 days

4

~~The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of concentration out of limits or inoperable SLC subsystems during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, an SLC subsystem is inoperable and that subsystem is subsequently returned the OPERABLE, the LCO may already have been not met for up to 7 days. This situation could lead to a total duration of 10 days (7 days in Condition B, followed by 3 days in Condition A), since initial failure of the LCO, to restore the SLC System. Then an SLC subsystem could be found inoperable again, and concentration could be restored to within limits. This could continue indefinitely.~~

7

BASES

ACTIONS (continued)

~~This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition A was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.~~ 7

B.1

If one SLC subsystem is inoperable for reasons other than Condition A, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the ATWS design basis ~~shutdown~~ function. However, the overall reliability is reduced because a single failure in the remaining OPERABLE subsystem could result in reduced SLC System shutdown capability. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the intended SLC System function and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive (CRD) System to shut down the plant. 2

~~The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of concentration out of limits or inoperable SLC subsystems during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, concentration is out of limits, and is subsequently returned to within limits, the LCO may already have been not met for up to 3 days. This situation could lead to a total duration of 10 days (3 days in Condition A, followed by 7 days in Condition B), since initial failure of the LCO, to restore the SLC System. Then concentration could be found out of limits again, and the SLC subsystem could be restored to OPERABLE. This could continue indefinitely.~~ 7

~~This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition B was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.~~

BASES

ACTIONS (continued)

C.1

If both SLC subsystems are inoperable for reasons other than Condition A, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor.

D.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, ~~and SR 3.1.7.3~~ 4

2

and

SR 3.1.7.1 ~~through~~ SR 3.1.7.~~3~~ are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution volume and temperature~~, including the temperature of the pump suction piping,~~ are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank ~~or in the pump suction piping~~. The temperature versus concentration curve of Figure 3.1.7-2 ensures that a ~~10~~°F margin will be maintained above the saturation temperature. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured parameters of volume and temperature. 4 4 4 2

5

INSERT 1

INSERT 2 4

2

INSERT 1

The volume of sodium pentaborate solution requirements in Figure 3.1.7-1 and Table 3.1.7-1 Equation 1 will ensure both the original design basis and the ATWS design basis are met. Figure 3.1.7-1 can only be used if the B-10 enrichment in the storage tank is ≥ 55.0 atom percent. If the volume requirement of Table 3.1.7-1 Equation 1 is utilized for verification of volume requirements the concentration requirements for the original design basis can also be considered to be met. However, to verify the ATWS design basis requirements are met, Table 3.1.7-1 Equation 2 must be used to verify the concentration of sodium pentaborate solution requirements are met.

4

INSERT 2

SR 3.1.7.3

SR 3.1.7.3 is a 24 hour Surveillance that requires the verification that the room temperature in the vicinity of the SLC pumps is within the solution temperature limits of Figure 3.1.7-2 or that the SLC pump suction lines heat tracing is OPERABLE. This Surveillance will help ensure that the proper borated solution temperature of the pump suction piping is maintained. Maintaining a minimum specified room temperature is important in ensuring that the boron remains in solution and does not precipitate out in the pump suction piping. The temperature versus concentration curve of Figure 3.1.7-2 ensures that a 5°F margin will be maintained above the saturation temperature. An acceptable alternate requirement is to verify the pump suction lines heat tracing is OPERABLE. The heat tracing is sized to maintain the pump suction above 70°F when the room temperature is 45°F. OPERABILITY of the heat tracing is confirmed by verifying the light associated with each controller is on, or by depressing the toggle switch and ensuring the light is on. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured room temperature.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.6 verifies that each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual ~~, power operated, and automatic~~ valves in the SLC System flow path provides assurance that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This Surveillance also does not apply to valves that are locked, sealed, or otherwise secured in position since they are verified to be in the correct position prior to locking, sealing, or securing. This verification of valve alignment does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensures correct valve positions. 4

SR 3.1.7.5

INSERT 3

sodium pentaborate

#

This Surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that ~~the proper concentration of~~ ~~boron~~ exists in the storage tank. ~~SR 3.1.7.5 must be performed anytime~~ ~~boron~~ ~~or water is added to the storage tank solution to determine that the~~ ~~boron~~ ~~solution concentration is within the specified limits. SR 3.1.7.5 must also be performed anytime the temperature is restored to within the limits of Figure 3.1.7-2, to ensure that no significant boron precipitation occurred.~~ The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of ~~boron~~ concentration between ~~s~~urveillances. 5 4 5

4

INSERT 3

The concentration of sodium pentaborate in solution required in Figure 3.1.7-1 will ensure the original design basis and the ATWS design basis are met. Figure 3.1.7-1 can only be used if the B-10 enrichment in the storage tank is $\geq$ 55.0 atom percent and as long as the flow rate requirements of SR 3.1.7.7 are met. Equation 2 of Table 3.1.7-1 ensures both the original design basis and ATWS design basis are satisfied. If the volume requirement of Equation 1 of Table 3.1.7-1 is utilized for verification of volume requirements the concentration requirements for the original design basis can also be considered to be met. However, to verify the ATWS requirements are met, Equation 2 of Table 3.1.7-1 must be used to verify the concentration of sodium pentaborate solution requirements are met.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.7

24

Demonstrating that each SLC System pump develops a flow rate 1275 ≥ ~~41.2~~ gpm at a discharge pressure ≥ ~~1190~~ psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. 4

tests

Such inservice ~~inspections~~ confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is ~~[~~ in accordance with the Inservice Testing Program ~~or 92 days]~~. 5 6

SR 3.1.7.8 and SR 3.1.7.9

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every ~~36~~ months at alternating ~~18~~ month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a ~~test tank~~ through one SLC subsystem and into the RPV. The ~~18~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the ~~18~~ month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

48 24 4

manually initiated 24 2 4

4 24

Demonstrating that all heat traced piping between the boron solution storage tank and the suction inlet to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping is unblocked is to pump from the storage tank to the test tank. 2

BASES

SURVEILLANCE REQUIREMENTS (continued)

and the SLC pump suction lines heat tracing is inoperable 3

24

The 18 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. This is especially true in light of the temperature verification of this piping required by SR 3.1.7.3. However, if, in performing SR 3.1.7.3, it is determined that the temperature of this piping has fallen below the specified minimum, SR 3.1.7.9 must be performed once within 24 hours after the piping temperature is restored to within the limits of Figure 3.1.7-2. 4

solution temperature

room 4 4

in the vicinity of the SLC pumps

SR 3.1.7.10

(laboratory analyses)

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Isotopic tests on the granular sodium pentaborate to verify the actual B-10 enrichment must be performed prior to addition to the SLC tank in order to ensure that the proper B-10 atom percentage is being used. 2

REFERENCES

1. 10 CFR 50.62.

6.6.1.1

U

2. FSAR, Section 4.2.3.4.3. 2 6

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.7 BASES, STANDBY LIQUID CONTROL (SLC) SYSTEM

1. Editorial change made for enhanced clarity.
2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases, which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.
3. Changes have been made to reflect the Specification.
4. Changes have been made to reflect those changes made to the Specification.
5. Typographical/grammatical error corrected.
6. The brackets have been removed and the proper plant specific information/value has been provided.
7. These changes are made consistent with TSTF-439, Rev. 2, which has been incorporated by the USNRC into Revision 3.1 of NUREG-1433.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.7, STANDBY LIQUID CONTROL (SLC) SYSTEM

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 8

ITS 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS | ITS

3.0 LIMITING CONDITIONS FOR OPERATION

F. Scram Discharge Volume

M.2 L.1

LCO 3.1.8 — 1. During reactor operation, the scram discharge volume vent and drain valves shall be operable, except as specified below.

Applicability

Add ACTIONS Note 1

ACTION A, ACTION B — 2. If any scam discharge volume vent or drain valve is made or found inoperable, the integrity of the scram discharge volume shall be maintained by either:

ACTION A — a. Verifying daily, for a period not to exceed 7 days, the operability of the redundant valve(s), or

L.2

ACTION A, ACTION B — b. Maintaining the inoperable valve(s), or the associated redundant valve(s), in the closed position.

Isolate the associated line — A.2

ACTIONS NOTE 2 — Periodically the inoperable and the redundant valve(s) may both be in the open position to allow draining the scram discharge volume.

Add proposed Required Action B.1 Completion Time — L.3

ACTION C — If a or b above cannot be met, at least all but one operable control rods (not including rods removed per specification 3.10.E or inoperable rods allowed by 3.3.A.2) shall be fully inserted within ten hours.

Be in MODE 3 — A.3

12 — L.4

G. Required Action

If Specifications 3.3.A through D above are not met, an orderly shutdown shall be initiated and have reactor in the cold shutdown condition within 24 hours.

See ITS 3.1.1, ITS 3.1.3, ITS 3.1.4, ITS 3.1.5, ITS 3.1.6, ITS 3.3.1.2, ITS 3.9.5

4.0 SURVEILLANCE REQUIREMENTS

F. Scram Discharge Volume

Add proposed SR 3.1.8.1 — M.1

SR 3.1.8.2 — The scram discharge volume vent and drain valves shall be cycled quarterly.

24 months — A.5

SR 3.1.8.3 — Once per operating cycle verify the scram discharge volume vent and drain valves close within 30 seconds after receipt of a reactor scram signal and open when the scram is reset.

"actual or simulated" — A.4

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DISCUSSION OF CHANGES
ITS 3.1.8, SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.3.F.2.b states, in part, "Maintaining the inoperable valves(s), or the associated redundant valve(s), in the closed position" if the inoperable valve is not restored to OPERABLE status in 7 days. ITS 3.1.8 Required Actions A.1 and B.1 state "Isolate the associated line." This changes the CTS by simplifying the Required Action by requiring isolation of the associated line instead of explicitly stating which valves to use to perform the isolation (i.e., inoperable valve(s) or the associated redundant valves(s)).

The purpose of CTS 3.3.F.2.b is to isolate the affected SDV vent or drain line. This change is acceptable since the proposed Required Action also requires isolation of the associated line, and the only valves capable of isolating the SDV vent and drain lines are the required SDV vent and drain valves. This change is designated as administrative because it does not result in any technical changes to the CTS.

A.3 CTS 3.3.F.2 states, in part, "If a or b above cannot be met, at least all but one operable control rods (not including rods removed per specification 3.10.E or inoperable rods allowed by 3.3.A.2) shall be fully inserted." ITS 3.1.8 ACTION C, under the same conditions requires the unit to be in MODE 3. This changes the CTS by more clearly defining the all rods in condition as MODE 3.

The purpose of CTS 3.3.F.2, when the requirements of CTS 3.3.F.2.a and b are not met, is to insert all operable control rods, which essentially ensures an inoperable SDV vent or drain valve cannot prevent a reactor scram. This change is acceptable because when the unit is in MODE 3, by definition, the reactor mode switch is in the shutdown condition and by design all OPERABLE rods will be inserted. The cross references to CTS 3.3.A.2 (Reactivity Margin-Stuck Control Rods) and CTS 3.10.E (Extended Core and Control Rod Drive Maintenance) are not necessary. CTS 3.3.A.2, which covers stuck control rods, is only applicable in MODES 1 and 2. CTS 3.10.E is only applicable during an outage. Therefore it is not necessary to include these cross references. This change is designated as administrative because it does not result in technical changes to the CTS.

A.4 CTS 3.3.F states, in part, "verify the scram discharge volume vent and drain valves close within 30 seconds after receipt of a reactor scram signal and open when the scram is reset." ITS SR 3.1.8.3 requires the same test however the proposed Surveillance states that the reactor scram signal may be an "actual or simulated" signal. This changes the CTS by clarifying that the reactor scram signal may be either an "actual or simulated" reactor scram signal.

DISCUSSION OF CHANGES
ITS 3.1.8, SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

The purpose of the test is to verify that the valves close and open on the specified signal. OPERABILITY is adequately demonstrated in either case since the SDV vent and drain valves cannot discriminate between "actual" or "simulated" signals. In addition, the CTS does not prohibit the signal from being an "actual" or "simulated" reactor scram signal. This change only clarifies the type of signal that may be used to perform the Surveillance Requirement. This change is designated as administrative because it does not result in any technical changes to the CTS.

A.5 CTS 4.3.F requires a SDV vent and drain valve test to be performed "Once per operating cycle." ITS SR 3.1.8.3 requires performance of an SDV vent and drain valve test every "24 months." This changes the CTS by changing the Frequency from "Once per operating cycle" to "24 months."

This change is acceptable because the current "operating cycle" is "24 months." In letter L-MT-04-036, from Thomas J. Palmisano (NMC) to the USNRC, dated June 30, 2004, NMC has proposed to extend the fuel cycle from 18 to 24 months and the same time has performed an evaluation in accordance with Generic Letter 91-04 to extend the unit Surveillance Requirements from 18 months to 24 months. CTS 4.3.F was included in this evaluation. This change is designated as administrative because it does not result in any technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 ITS SR 3.1.8.1 requires the verification that each SDV vent and drain valve is open. A Note is included that states that this Surveillance is not required to be met on vent and drain valves closed during performance of SR 3.1.8.2. The CTS does not contain a similar requirement. This changes the CTS by adding a new Surveillance Requirement for the SDV vent and drain valves.

This change is acceptable because it helps to ensure the SDV is capable of performing its intended safety function. During normal operation, the SDV vent and drain valves should be in the open position to allow for drainage of the SDV piping. The Surveillance includes a Note to allow the SDV vent and drain valves to be cycled in accordance with ITS SR 3.1.8.2 without declaring the associated valves inoperable. This verification gives additional confidence that the SDV is available to receive and contain all the water discharge by the control rod drives during a scram. This change is designated as more restrictive because it adds a Surveillance Requirement that does not appear in the CTS.

M.2 CTS 3.3.F requires the scram discharge volume vent and drain valve requirements to be met in the "reactor operation" condition. ITS LCO 3.1.8 is Applicable in MODES 1 and 2. This changes the CTS by requiring the scram discharge volume vent and drain valve requirements to be met in MODE 2 $\leq$ 1% RATED THERMAL POWER (RTP).

The purpose of CTS 3.3.F is to ensure the scram discharge volume vent and drain valve requirements are met to ensure the negative scram reactivity is consistent with those values assumed in the accident and transient analysis.

DISCUSSION OF CHANGES
ITS 3.1.8, SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

This change expands the Applicability to require the scram discharge volume vent and drain valve requirements to be met at all times when in MODE 2, instead of when > 1% RTP (the CTS 1.0.O definition states that Power Operation is when reactor power is > 1% RTP). This change is acceptable since the control rods must be capable of properly scramming in MODE 2 because the reactor is critical or control rods are withdrawn (thus the need exists for the scram discharge volume vent and drain valves to be OPERABLE). This change is designated as more restrictive because the LCO will be applicable under more reactor conditions.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* CTS 3.3.F.2.a allows 7 days of continuous operation with any number of SDV drain or vent valves inoperable as long as the redundant valve (i.e., the one in the same line) is verified to be OPERABLE on a daily basis. After the 7 day period, CTS 3.3.F.2.b requires that either the inoperable valve(s) or the associated redundant valve(s) be closed. However, if one valve has been inoperable for greater than 7 days and the valve or its redundant valve is closed, and another valve in a different line becomes inoperable, the CTS does not allow a separate 7 day time to restore the valve; the second inoperable valve or its redundant valve must be closed immediately in order to meet the requirements of CTS 3.3.F.2.b. ITS 3.1.8 ACTIONS are modified by a Note 1 that states "Separate Condition entry is allowed for each SDV vent and drain line." ITS 3.1.8 ACTION A covers inoperabilities for one or more SDV vent or drain lines with one valve inoperable. ITS 3.1.8 ACTION B covers inoperabilities for one or more SDV vent or drain lines with both valves inoperable. This changes the CTS by allowing separate Condition entry for each inoperable SDV vent or drain line. That is, under the same scenario described above, the second inoperable valve will get a 7 day restoration time before the associated line must be isolated. Other modifications associated with CTS 3.3.F.2.a and CTS 3.3.F.2.b are discussed in DOCs A.2, L.2, and L.3.

The purpose of CTS 3.3.F.2.a is to allow 7 days of operation when any number of SDV vent or drain valves are inoperable as long as the associated redundant valve on the same line is operable. The purpose of CTS 3.3.F.2.b is to require immediate closure of the inoperable valves if both valves on a SDV vent or drain line are inoperable and to also require isolation of the affected penetration after 7 days of operation with any SDV vent or drain valve inoperable. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to

DISCUSSION OF CHANGES
ITS 3.1.8, SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period. This change will allow separate Condition entry for each SDV vent and drain line. This change will effectively allow 7 days to isolate the affected line when one valve in the line is discovered to be inoperable. This is acceptable since the since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SDV line. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SDV lines are governed by subsequent Condition entry and application of associated Required Actions. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 4 – Relaxation of Required Action)* When any scram discharge volume vent or drain valve is made or found inoperable and the associated line is not isolated, CTS 3.3.F.2.a requires daily verification of the OPERABILITY of the redundant valve(s). ITS 3.1.8 ACTION A covers the condition when one SDV vent or drain valve is inoperable in one or more SDV vent or drain lines, but does not require daily verification of the OPERABILITY of the redundant valve in the associated line if the line is not isolated. This changes the CTS by deleting the requirement to verify the OPERABILITY of the redundant valve(s) on a daily basis if the associated line is not isolated.

The purpose of CTS 3.3.F.2.a is to provide compensatory actions for inoperable SDV vent or drain valves. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the repair period. This change deletes the requirement to verify on a daily basis the OPERABILITY of the redundant valve. ITS SR 3.1.8.2 requires that each SDV vent and drain valve to be cycled to the fully closed and fully open position every 31 days. ITS SR 3.1.8.3 requires the verification that each valves actuates as required on a scram signal every 24 months. These Surveillances and associated Frequencies are considered sufficient to determine whether or not a SDV vent or drain valve is OPERABLE. As stated in the Bases of ITS SR 3.0.3, it is recognized that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. This change is acceptable since the normal Surveillances and associated Frequencies are considered acceptable with respect to determining the status of a SDV vent or drain valve. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

DISCUSSION OF CHANGES
ITS 3.1.8, SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

L.3 *(Category 3 – Relaxation of Completion Time)* When any scram discharge volume vent or drain valve is made or found inoperable, CTS 3.3.F.2.a allows, for a period not to exceed 7 days, the associated line to remain unisolated provided the redundant valve in the line is OPERABLE. If both valves in a SDV line are inoperable, CTS 3.3.F.2.b requires "maintaining" the inoperable valve(s) or the associated redundant valve(s) in the closed position. This effectively means that if both valves in a SDV line are inoperable, the line must be isolated immediately. ITS 3.1.8 ACTION B covers the condition when both valves are inoperable in one or more SDV vent or drain lines. ITS 3.1.8 Required Action B.1 requires isolation of the associated line within 8 hours. This changes the CTS by allowing 8 hours to isolate a vent or drain line in lieu of requiring it to be isolated immediately when both valves are determined to be inoperable.

The purpose of CTS 3.3.F.2 is to provide compensatory actions for inoperable SDV vent or drain valves. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. This change extends the time from immediately to 8 hours to isolate a SDV vent or drain line when it is determined that both valves associated with the same SDV vent or drain line are inoperable. If both valves in a line are inoperable, the line must be isolated to contain the reactor coolant during a scram. The 8 hour Completion Time to isolate the line is acceptable based on the low probability of a scram occurring while the line is not isolated and unlikelihood of significant CRD seal leakage. This change is designated as less restrictive because additional time is allowed to isolate the SDV line than was allowed in the CTS.

L.4 *(Category 3 – Relaxation of Completion Time)* CTS 3.3.F.2 requires the insertion of all OPERABLE control rods within ten hours if the compensatory actions of CTS 3.3.F.2.a and b cannot be met. ITS 3.1.8 ACTION C requires the unit to be in MODE 3 in 12 hours. This change increases the time to insert all OPERABLE control rods (i.e., to be in MODE 3 as discussed in DOC A.3) from 10 hours to 12 hours.

The purpose of the action in CTS 3.3.F.2 is to insert all OPERABLE control rods (which ensures an inoperable SDV vent or drain valve cannot prevent a reactor scram) in an acceptable time frame. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the allowed Completion Time. The Completion Time of 12 hours is needed to give the operator sufficient time to accomplish an orderly power reduction without challenging unit systems. This proposed Completion Time is consistent with the Completion Times to achieve MODE 3 in all other ITS Specifications. The inoperabilities of SDV vent and drain valves should not require the unit to reach MODE 3 any faster than other Specification requiring entry into this same MODE. This change is designated as less restrictive because additional time is allowed to insert all OPERABLE control rods than was allowed in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS 3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

3.3.F.1 LCO 3.1.8 Each SDV vent and drain valve shall be OPERABLE.

3.3.F.1 APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTES-----

DOC L.1 1. Separate Condition entry is allowed for each SDV vent and drain line.

3.3.F.2.b 2. An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.3.F.2, 3.3.F.2.a, 3.3.F.2.b	A. One or more SDV vent or drain lines with one valve inoperable.	A.1 Isolate the associated line.	7 days
3.3.F.2, 3.3.F.2.b	B. One or more SDV vent or drain lines with both valves inoperable.	B.1 Isolate the associated line.	8 hours
3.3.F.2	C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

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CTS SURVEILLANCE REQUIREMENTS

CTS		SURVEILLANCE	FREQUENCY	
DOC M.1	SR 3.1.8.1	-----NOTE----- Not required to be met on vent and drain valves closed during performance of SR 3.1.8.2. ---------- Verify each SDV vent and drain valve is open.	31 days	
4.3.F	SR 3.1.8.2	Cycle each SDV vent and drain valve to the fully closed and fully open position.	92 days	
4.3.F	SR 3.1.8.3	Verify each SDV vent and drain valve: a. Closes in ≤ ~~[60]~~ 30 seconds after receipt of an actual or simulated scram signal ; and b. Opens when the actual or simulated scram signal is reset.	~~[18]~~ 24 months	1 1 2

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.8, SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

1. The brackets are removed and the proper plant specific information/value is provided.

2. These punctuation corrections have been made consistent with the Writer's Guide for the Standard Technical Specifications, NEI 01-03, Section 5.1.3.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving ~~approximately one~~ half of the control rod drive (CRD) discharges. Each ~~The two~~ instrument volume~~s~~ is ~~are~~ connected to a ~~common~~ drain line with two valves in series for a total of four drain valves. Each header is connected to a ~~common~~ vent line with two valves in series for a total of four vent valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

1

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 2) and

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 100 (Ref. 2), and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in the instrument volume exceeds a specified setpoint. The setpoint is chosen so that all control rods are inserted before the SDV has insufficient volume to accept a full scram.

SDV vent and drain valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The OPERABILITY of all SDV vent and drain valves ensures that the SDV vent and drain valves will close during a scram to contain reactor water discharged to the SDV piping. Since the vent and drain lines are provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to open on scram reset to ensure that a path is available for the SDV piping to drain freely at other times.

APPLICABILITY

a

In MODES 1 and 2, scram may be required; therefore, the SDV vent and drain valves must be OPERABLE. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. ~~This provides adequate controls to ensure that only a single control rod can be withdrawn.~~ Also, during MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Therefore, the SDV vent and drain valves are not required to be OPERABLE in these MODES since the reactor is subcritical and only one rod may be withdrawn and subject to scram. 3 2

ACTIONS

The ACTIONS table is modified by Note 1 indicating that a separate Condition entry is allowed for each SDV vent and drain line. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SDV line. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SDV lines are governed by subsequent Condition entry and application of associated Required Actions. 3

INSERT 1

When a line is isolated, the potential for an inadvertent scram due to high SDV level is increased. During these periods, the line may be unisolated under administrative control. This allows any accumulated water in the line to be drained, to preclude a reactor scram on SDV high level. This is acceptable since the administrative controls ensure the valve can be closed quickly, by a dedicated operator, if a scram occurs with the valve open. 2

A.1

When one SDV vent or drain valve is inoperable in one or more lines, the associated line must be isolated to contain the reactor coolant during a scram. The 7 day Completion Time is reasonable, given the level of redundancy in the line~~s~~ and the low probability of a scram occurring while the valve~~s~~ ~~are~~ is inoperable and the line is not isolated. The SDV is still isolable since the redundant valve in the affected line is OPERABLE. During these periods, the single failure criterion may not be preserved, and a higher risk exists to allow reactor water out of the primary system during a scram. a 3 3

2

INSERT 1

The ACTIONS Table is modified by a second Note stating that an isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

BASES

ACTIONS (continued)

B.1

If both valves in a line are inoperable, the line must be isolated to contain the reactor coolant during a scram. 2

The 8 hour Completion Time to isolate the line is based on the low probability of a scram occurring while the line is not isolated and unlikelihood of significant CRD seal leakage.

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to ~~at least~~ MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. 2

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

During normal operation, the SDV vent and drain valves should be in the open position (except when performing SR 3.1.8.2) to allow for drainage of the SDV piping. Verifying that each valve is in the open position ensures that the SDV vent and drain valves will perform their intended functions during normal operation. This SR does not require any testing or valve manipulation; rather, it involves verification that the valves are in the correct position.

The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation, which ensure correct valve positions.

SR 3.1.8.2

During a scram, the SDV vent and drain valves should close to contain the reactor water discharged to the SDV piping. Cycling each valve through its complete range of motion (closed and open) ensures that the valve will function properly during a scram. The 92 day Frequency is based on operating experience and takes into account the level of redundancy in the system design.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.3

SR 3.1.8.3 is an integrated test of the SDV vent and drain valves to verify total system performance. After receipt of a simulated or actual scram signal, the closure of the SDV vent and drain valves is verified. The closure time of [60] seconds after receipt of a scram signal is based on the bounding leakage case evaluated in the accident analysis (Ref. [2]). Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3 overlap this Surveillance to provide complete testing of the assumed safety function. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the [18] month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

30 4 3 1 , "Control Rod OPERABILITY," 2 24 4 24 4

REFERENCES U

1. [F]SAR, Section [4.2.3.2.2.3]. 1 5 3.5.3.3.3.5

2. 10 CFR 100.

3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.

JUSTIFICATION FOR DEVIATIONS
ITS 3.1.8 BASES, SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases, which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. Typographical/grammatical error corrected.

4. Changes have been made to reflect those changes made to the Specification.

5. The brackets are removed and the proper plant specific information/value is provided.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.1.8, SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

There are no specific NSHC discussions for this Specification.

ATTACHMENT 9

Relocated/Deleted Current Technical Specifications

CTS 3/4.3.B.2, Control Rod Drive Housing Support System

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	(b) when the rod is withdrawn the first time subsequent to each refueling outage, observe discernible response of the nuclear instrumentation. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is critical shall be performed to observe nuclear instrumentation response.
2. ~~The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all operable control rods are fully inserted and Specification 3.3.A.1 is met.~~	2. ~~The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.~~
3.(a) Control rod withdrawal sequences shall be established so that the maximum calculated reactivity that could be added by dropout of any increment of any one control blade will not make the core more than 1.3% Δk supercritical.	3.(a) To consider the rod worth minimizer operable, the following steps must be performed: (i) The control rod withdrawal sequence for the rod worth minimizer computer shall be verified as correct. (ii) The rod worth minimizer computer on-line diagnostic test shall be successfully completed. (iii) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

See ITS 3.1.3

L.1

See ITS 3.3.2.1

See ITS 3.1.6

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DISCUSSION OF CHANGES
CTS 3/4.3.B.2, CONTROL ROD DRIVE HOUSING SUPPORT SYSTEM

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 1 – Relaxation of LCO Requirement)* CTS 3/4.3.B.2 requires the control rod drive housing support system to be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all operable control rods are fully inserted and Specification 3.3.A.1 is met. CTS 4.3.B.2 requires the control rod drive housing support system to be inspected after reassembly and the results of the inspection recorded. ITS 3.1 does not include the requirements for the control rod drive housing support system. This changes the CTS by deleting the explicit control rod drive housing support system requirements from the Technical Specifications.

The purpose of CTS 3/4.3.B.2 is to ensure that the control rod drive housing support system is operable when control rods are withdrawn from the core. This change is acceptable because the LCO requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. The CTS 3/4.3.B.2 requirement for the control rod drive housing support to be in place is included in the OPERABILITY requirements for control rods. Plant configuration management provides adequate controls to assure the control rod drive housing support is in place. The current Technical Specifications require the control rod drive housing support system to be inspected after reassembly and the results of the inspection recorded. This current Technical Specifications requirement verifies that the control rod drive housing support is in place for reactor operation in MODES 1, 2, and 3. Post-maintenance inspections conducted through plant configuration management control have the same function as the current Technical Specifications requirement. Since work is not normally performed on the control rod drive housing support at power, and checks on its installation are not made at power there is no current requirement to verify control rod drive housing support installation in power operating conditions. Therefore, the deletion of this current

DISCUSSION OF CHANGES
CTS 3/4.3.B.2, CONTROL ROD DRIVE HOUSING SUPPORT SYSTEM

Technical Specifications is acceptable based on use of plant configuration management control to ensure proper control rod drive housing support system installation. This change is designated as a less restrictive change because a requirement is being removed from the Technical Specifications.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS 3/4.3.B.2, CONTROL ROD DRIVE HOUSING SUPPORT SYSTEM

There are no specific NSHC discussions for this Specification.

Summary of Changes
ITS Section 3.2

Change Description	Affected Pages
The changes described in the NMC response to Question 200601201447 have been made. Minor grammatical correction to the ITS Bases has been made.	Page 19 of 73

ATTACHMENT 1

VOLUME 7

MONTICELLO
IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

ITS SECTION 3.2
POWER DISTRIBUTION LIMITS

Revision 1

LIST OF ATTACHMENTS

ATTACHMENT 1

ITS 3.2.1, AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3.11 REACTOR FUEL ASSEMBLIES	4.11 REACTOR FUEL ASSEMBLIES
Applicability: The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.	Applicability: The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.
Objective: The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.	Objective: The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.
Specifications:	Specifications:
A. Average Planar Linear Heat Generation Rate (APLHGR)	A. Average Planar Linear Heat Generation Rate (APLHGR)

LCO 3.2.1

During two ~~recirculation loop~~ power operation, the APLHGR for each type of ~~fuel as a function of average planar exposure~~ shall not exceed the applicable limiting values specified in the Core Operating Limits Report. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) provided in the Core Operating Limits Report.

During one ~~recirculation loop~~ power operation, the APLHGR limiting condition for operation for each type of fuel shall not exceed the most limiting of:

a. The above values multiplied by 0.80 for GE11 and GE12 fuel and 0.90 for GE14 fuel, or

b. The above values multiplied by the appropriate flow and power dependent correction factors provided in the Core Operating Limits Report.

LA.1 A.2 LA.2 LA.1 A.2

SR 3.2.1.1

The APLHGR for each type ~~of fuel~~ as a function of average ~~planar~~ exposure shall be determined daily during reactor operation at ≥25% rated thermal power.

LA.1

Add proposed SR 3.2.1.1 first Frequency

L.1

A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
ACTION A: If at any time during power operation, it is determined that the APLHGR limiting condition for operation is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. (LA.3) Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. (A.3) If the APLHGR is not returned to within the prescribed limits within two hours, ACTION B: reduce thermal power to less than 25% within the next four hours.	
B. <u>Linear Heat Generation Rate (LHGR)</u> During power operation, the LHGR shall be less than or equal to the limits specified in the Core Operating Limits Report. If at any time during operation it is determined that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. If the LHGR is not returned to within the prescribed limits within 2 hours, reduce thermal power to less than 25% within the next 4 hours.	B. <u>Linear Heat Generation Rate (LHGR)</u> The LHGR shall be checked daily during reactor operation at ≥25% of rated thermal power.

See ITS 3.2.3

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DISCUSSION OF CHANGES
ITS 3.2.1, AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.11.A states that the APLHGR should not exceed limits during "power operation," which is defined in CTS 1.0.O as "above 1% rated thermal power." However, CTS 3.11.A only states to reduce thermal power to "less than 25%" if the APLHGR LCO is being exceeded and the APLHGRs are not returned to within limits within the specified time. ITS LCO 3.2.1 is applicable at THERMAL POWER ≥ 25% RTP. ITS 3.2.1 ACTION B requires a THERMAL POWER reduction to < 25% RTP if the APLHGR(s) are not restored to within limits within the specified time limit of ACTION A. This changes the CTS by changing the Applicability from > 1% rated thermal power to ≥ 25% RTP.

The purpose of the CTS 3.11.A is to ensure the APLHGRs are within limits when required. This changes the CTS by changing the Applicability from "power operation" to "≥ 25% RTP." This change is acceptable because at THERMAL POWER levels < 25% RTP the reactor is operating with substantial margin to the APLHGR limits. For this reason there is no need to monitor APLHGRs when THERMAL POWER is < 25% RTP. This is also consistent with the Surveillance Frequency in CTS 4.11.A, which states to monitor APLHGR at ≥ 25% rated thermal power. This change simply aligns the Applicability with the CTS default action and Surveillance Frequency, and is therefore considered administrative. Therefore, this change is considered a presentation preference change only and, as such, is considered an administrative change.

A.3 CTS 3.11.A states "Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits." ITS 3.2.1 does not include this statement. This changes the CTS by deleting this statement.

The purpose of this CTS 3.11.A statement is to identify the importance of monitoring the APLGHRs to verify they are restored to prescribed limits. After they are within limits, it is obvious that the action can be exited. This change is acceptable because ITS LCO 3.0.1 and LCO 3.0.2 have been added to the TS as indicated in the Discussion of Changes for ITS Section 3.0. ITS LCO 3.0.1 states "LCOs shall be met during the MODES or other specified conditions in the Applicability," and LCO 3.0.2 states "Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met" and "If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required." The CTS 3.11.A guidance is provided in the ITS generic guidelines of LCO 3.0.1 and LCO 3.0.2. In addition, the only way to confirm the APLHGRs have been restored to within limits is to perform a Surveillance; thus, it is not necessary to

DISCUSSION OF CHANGES
ITS 3.2.1, AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

be specifically stated. Therefore, this change is considered a presentation preference change only and, as such, is considered an administrative change.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.11.A specifies the limits for APLHGRs for "two loop" and "one loop" operation. For two loop operation, the APLGHR limits are specified "for each type of fuel as a function of average planar exposure." For one loop operation, the APLGHR limits are specified "for each type of fuel" and shall not exceed "the most limiting of a. The above values multiplied by 0.80 for GE11 and GE12 fuel and 0.90 for GE14 fuel, or b. The above values multiplied by the appropriate flow and power dependent correction factors provided in the Core Operating Limits Report." In addition CTS 4.11.A states the APLHGR "for each type of fuel as a function of average planar exposure" shall be determined. ITS 3.2.1 states "All APLHGRs shall be less than or equal to the limits specified in the COLR." ITS SR 3.2.1.1 requires verification of all APLGHRs are less than or equal to the limits specified in the COLR. This changes the CTS by relocating the details that the APLHGRs limits are specified for "one" and "two" loop operation, that the two loop APLHGR limits are specified "for each type of fuel as a function of average planar exposure," and that the single loop APLHGRs limits "for each type of fuel" shall not exceed "the most limiting of a. The above values multiplied by 0.80 for GE11 and GE12 fuel and 0.90 for GE14 fuel, or b. The above values multiplied by the appropriate flow and power dependent correction factors provided in the Core Operating Limits Report" to the Bases.

The removal of these details for evaluating APLGHR Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS LCO 3.2.1 still retains the requirement that "All APLHGRs shall be less than or equal to the limits specified in the COLR" and ITS SR 3.2.1.1 requires verification that "all APLHGRs are less than or equal to the limits specified in the COLR." Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail

DISCUSSION OF CHANGES
ITS 3.2.1, AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.2 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 3.11.A states "When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) provided in the Core Operating Limits Report." ITS LCO 3.2.1 states "All APLHGRs shall be less than or equal to the limits specified in the COLR." This changes the CTS by relocating the hand calculation APLHGR limits to the COLR.

The removal of these cycle-specific parameter limits from the Technical Specifications and their relocation into the COLR is acceptable because these limits are developed or utilized under NRC-approved methodologies. The NRC documented in Generic Letter 88-16, Removal of Cycle-Specific Parameter Limits From the Technical Specifications, that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements and Surveillances that verify that the cycle-specific parameter limits are being met. ITS 3.2.1 LCO requires, "All APLHGRs shall be less than or equal to the limits specified in the COLR," and ITS SR 3.2.1.1 requires verification that "all APLHGRs are less than or equal to the limits specified in the COLR." Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.3, "Core Operating Limits Report." ITS 5.6.3 ensures the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits) of the safety analysis are met. This change is designated as a less restrictive removal of detail change because information relating to cycle-specific parameter limits is being removed from the Technical Specifications.

LA.3 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.11.A states that if at any time during power operation it is determined that the APLHGR limiting condition for operation is being exceeded, "action shall be initiated within 15 minutes to restore operation to within the prescribed limits." ITS 3.2.1 does not include this 15 minute action. This changes the CTS by relocating the procedural detail that "action shall be initiated within 15 minutes to restore operation to within the prescribed limits" to the Bases in the form of a discussion that "prompt action should be taken to restore the APLHGR(s) to within the required limits."

The removal of this detail for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to restore the APLHGRs to within limits in 2 hours, consistent with the CTS actions. Also, this change is acceptable because this type of procedural detail will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly

DISCUSSION OF CHANGES
ITS 3.2.1, AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

controlled. This change is designated as a less restrictive removal of detail change because a procedural detail for meeting Technical Specification requirements is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.11.A requires the APLHGR to be determined daily during reactor operation at ≥ 25% rated thermal power. ITS SR 3.2.1.1 requires the same verification "once within 12 hours after ≥ 25% RTP and 24 hours thereafter." This changes the CTS by allowing the reactor to reach and exceed a THERMAL POWER level of 25% RTP without completing the Surveillance.

The purpose of CTS 4.11.A is to ensure all APLHGRs are within limits before THERMAL POWER is ≥ 25% RTP. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of fuel reliability. This change allows the plant to increase THERMAL POWER ≥ 25% RTP without completing the Surveillance. However, after 25% RTP is achieved the verification must be performed within 12 hours and every 24 hours thereafter. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

3.11.A LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

3.11.A APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.11.A	A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	2 hours
3.11.A	B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY
4.11.A	SR 3.2.1.1	Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

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JUSTIFICATION FOR DEVIATIONS
ITS 3.2.1, AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

None

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Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during anticipated operational occurrences (AOOs) and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46. node 4

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), anticipated operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, 5, 6, ~~and 7~~. 7, 8, 9, and 10 1

7, 8, 9, and 10

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during AOOs for operation with LHGRs up to the operating limit LHGR. APLHGR ~~limits~~ are equivalent to ~~the LHGR~~ limit ~~for~~ each fuel rod divi~~ded by~~ the local peaking fac~~tor of~~ the fuel assembly. 1 APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting AOOs (Refs. ~~5, 6, and 7~~). 1 Flow dependent APLHGR limits are determined using the three dimensional 11 BWR simulator code (Ref. ~~8~~) to analyze slow flow runout transients. 1 The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and AOOs. A complete discussion of the analysis code is provided in Reference ~~9~~. 12 1

All changes are "1" unless otherwise noted

BASES

APPLICABLE SAFETY ANALYSES (continued)

LOCA analyses are then performed to ensure that the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference 10. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. The APLHGR limits specified ~~are~~ equivalent ~~to~~ the LHGR of the highest powered fuel ~~rod~~ assumed in the LOCA analysis divided by its ~~local~~ peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

13 14 INSERT 1

For single recirculation loop operation, the MAPFAC multiplier is limited to a maximum of ~~0.75~~ (Ref. 5). This maximum limit is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

The APLHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The APLHGR limits specified in the COLR for each type of fuel as a function of average planar exposure are the result of the fuel design, DBA, and transient analyses. For two recirculation loops operating, the limit is determined by multiplying the smaller of the $MAPFAC_p$ and $MAPFAC_f$ factors times the exposure dependent APLHGR limits. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by the smaller of either $MAPFAC_p$, $MAPFAC_f$, ~~and~~ or ~~0.75~~ 0.80 and 0.90, where ~~0.75~~ 0.80 and 0.90 ha~~s~~ve ~~been determined by a specific single recirculation loop analysis (Ref. 5).~~

INSERT 2 3

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 7 NEDC-30492-P) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 25% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

10

1

INSERT 1

0.80 for GE11 and GE12 fuel and 0.90 for GE14 fuel

1

INSERT 2

0.80 for GE11 and GE12 fuel and 0.90 for GE14 fuel

BASES

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to ~~in~~ a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems. 3

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NED~~O~~E-24011-P-A "General Electric Standard Application for Reactor Fuel" ~~(latest approved version)~~ (revision specified in Specification 5.6.3) 3 4
2. U~~F~~SAR, Chapter ~~[4]~~ 3 1 2
3. U~~F~~SAR, ~~Chapter [6]~~ Section 6.2.6
4. U~~F~~SAR, ~~Chapter [15]~~ Section 14.7.2
5. USAR, Section 14.3. 1
6. USAR, Chapter 14A.
~~5. [Plant specific single loop operation].~~ INSERT 3 2
~~6. [Plant specific load line limit analysis].~~ INSERT 4

2

INSERT 3

7. NEDE-23785-P (A), Revision 1, "The GESTR-LOCA and SAFER Models for Evaluation of the Loss-of-Coolant Accident (Volume III), SAFER/GESTR Application Methodology," October 1984.

2

INSERT 4

8. NEDC-30515, "GE BWR Extended Load Line Limit Analysis for Monticello Nuclear Generating Plant, Cycle 11," March 1984.

9. NEDC-31849P, including Supplement 1, "Maximum Extended Load Line Limit Analysis for Monticello Nuclear Generating Plant Cycle 15," June 1992.

BASES

REFERENCES (continued)

INSERT 5

7. [Plant Specific Average Power ~~Range~~ Monitor, Rod Block Monitor and Technical ~~Specification~~ Improvements (ARTS) Program]. 2

1 11 ~~8~~. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

12 ~~9~~. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

INSERT 6

10. [Plant specific ~~loss of coolant~~ accident analysis]. 2

2 **INSERT 5**

10. NEDC-30492-P, "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Monticello Nuclear Generating Plant," April 1984.

2 **INSERT 6**

13. GE-NE-187-02-0392, "Monticello Nuclear Generating Plant SAFER/GESTR-LOCA Analysis Basis Documentation," July 1993.

14. Supplemental Reload Licensing Report for Monticello Nuclear Generation Plant (version specified in the COLR).

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.1 BASES, AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases, which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The brackets are removed and the proper plant specific information/value is provided.

3. Typographical/grammatical error corrected.

4. Editorial change made for clarity.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.1, AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 2

ITS 3.2.2, MINIMUM CRITICAL POWER RATIO (MCPR)

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS | ITS

3.0 LIMITING CONDITIONS FOR OPERATION

C. Minimum Critical Power Ratio (MCPR)

LCO 3.2.2

All MCPRs shall be greater than or equal to the MCPR Operating limits provided in the Core Operating Limits Report.

Add proposed Applicability — A.2

ACTION A

If at any time during operation it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two hours, reduce thermal power to less than 25% within the next four hours.

ACTION B

LA.1

A.3

4.0 SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

Add proposed ITS SR 3.2.2.1 first Frequency — L.1

SR 3.2.2.1

MCPR shall be determined daily during reactor power operation at ≥25% rated thermal power and following ~~any change in power level or distribution~~ which has the potential of bringing the core to its operating MCPR Limit.

L.2

Add proposed ITS SR 3.2.2.2 — M.1

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DISCUSSION OF CHANGES
ITS 3.2.2, MINIMUM CRITICAL POWER RATIO (MCPR)

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.11.C does not state when the MCPR LCO is required to be met, however CTS 3.11.C states "reduce thermal power to less than 25%" if the limiting value for MCPR is being exceeded and the MCPR is not returned to within limits within the specified time. ITS LCO 3.2.2 is applicable at THERMAL POWER ≥ 25% RTP. ITS 3.2.2 ACTION B requires a THERMAL POWER reduction to < 25% RTP if the MCPR(s) are not restored to within limits within specified time limit of ACTION A. This changes the CTS by clearly specifying the Applicability as ≥ 25% RTP.

The purpose of the CTS 3.11.C is to ensure the MCPRs are within limits when required. This changes the CTS by adding the explicit Applicability of "THERMAL POWER ≥ 25% RTP." This change is acceptable because at THERMAL POWER levels < 25% RTP the reactor is operating with substantial margin to the MCPR limits. For this reason there is no need to monitor MCPRs when THERMAL POWER is < 25% RTP. This is also consistent with the Surveillance Frequency in CTS 4.11.C, which states to monitor MCPR at ≥ 25% rated thermal power. This change states the Applicability consistent with the CTS default action and Surveillance Frequency. Therefore, this change is considered a presentation preference change only and, as such, is considered an administrative change.

A.3 CTS 3.11.C states "Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits." ITS 3.2.2 does not include this statement. This changes the CTS by deleting this statement.

The purpose of this CTS 3.11.C statement is to identify the importance of monitoring the MCPRs to verify they are restored to prescribed limits. After the MCPRs are within limits, it is obvious that the action can be exited. This change is acceptable because ITS LCO 3.0.1 and LCO 3.0.2 have been added to the Technical Specifications as described in the Discussion of Changes for ITS Section 3.0. ITS LCO 3.0.1 states "LCOs shall be met during the MODES or other specified conditions in the Applicability," and LCO 3.0.2 states "Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met" and "If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required." The CTS 3.11.C guidance is provided in the ITS generic guidelines of LCO 3.0.1 and LCO 3.0.2. In addition, the only way to confirm the MCPRs have been restored to within limits and the LCO is being met is to perform a Surveillance; thus, it is not necessary to be specifically stated.

Therefore, this change is considered a presentation preference change only and, as such, is considered an administrative change.

MORE RESTRICTIVE CHANGES

M.1 CTS 4.11.C does not specify a Surveillance Requirement to determine the MCPR limits after completion of scram time testing. ITS SR 3.2.2.2 requires the determination of the MCPR limits once within 72 hours after each completion of SR 3.1.4.1, once within 72 hours after each completion of SR 3.1.4.2, and once within 72 hours after each completion of SR 3.1.4.4 (scram time testing Surveillances). This changes the CTS by adding ITS SR 3.2.2.2 to the Technical Specifications.

The purpose of ITS SR 3.2.2.2 is to determine the MCPR limits after performance of the scram time tests, since scram times can affect the MCPR limit. This change is acceptable because the transient analysis is allowed to take credit for conservatism in the scram speed performance, thus it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. ITS SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4, "Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter τ must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle. This change is more restrictive because it adds a Surveillance Requirement that prescribes explicit requirements to determine MCPR limits at the specified times.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.11.C states that if at any time during power operation it is determined that the limiting value for MCPR is being exceeded, "action shall be initiated within 15 minutes to restore operation to within the prescribed limits." ITS 3.2.2 does not include this 15 minute action. This changes the CTS by relocating the procedural detail that "action shall be initiated within 15 minutes to restore operation to within the prescribed limits" to the Bases in the form of a discussion that "prompt action should be taken to restore the MCPR(s) to within the required limits."

DISCUSSION OF CHANGES
ITS 3.2.2, MINIMUM CRITICAL POWER RATIO (MCPR)

The removal of this detail for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to restore the MCPRs to within limits in 2 hours, consistent with the CTS actions. Also, this change is acceptable because this type of procedural detail will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because a procedural detail for meeting Technical Specification requirements is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.11.C requires the MCPR to be determined daily during reactor operation at ≥ 25% rated thermal power. ITS SR 3.2.2.1 requires the same verification "once within 12 hours after ≥ 25% RTP and 24 hours thereafter." This changes the CTS by allowing the reactor to reach and exceed a THERMAL POWER level of 25% RTP without completing the Surveillance.

The purpose of CTS 4.11.C is to ensure all MCPRs are within limits before THERMAL POWER is ≥ 25% RTP. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of fuel reliability. This change allows the plant to increase THERMAL POWER ≥ 25% RTP without completing the Surveillance. However, after 25% RTP is achieved the verification must be performed within 12 hours and every 24 hours thereafter. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.2 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.11.C states MCPR shall be determined daily and "following any change in power level or distribution which has the potential of bringing the core to its operating MCPR." ITS SR 3.2.2.1 requires a similar daily verification, but does not include the additional Frequency based on a change in power level or distribution. This changes the CTS by deleting the requirement to verify MCPRs are within limits "following any change in power level or distribution which has the potential of bringing the core to its operating MCPR."

The purpose of the above described CTS 4.11.C Surveillance Frequency is to ensure MCPR is within limits when there is a potential for bringing the core to its operating MCPR limit. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of fuel reliability. This condition is unlikely and the Surveillance would seldom be required. Therefore, the Surveillance Frequency has been deleted. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

3.11.C LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

3.11.C APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.11.C	A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
3.11.C	B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.11.C	SR 3.2.2.1	Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
DOC M.1	SR 3.2.2.2	Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 AND Once within 72 hours after each completion of SR 3.1.4.2 AND Once within 72 hours after each completion of SR 3.1.4.4

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.2, MINIMUM CRITICAL POWER RATIO (MCPR)

None

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion. 3 3 3

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, 7, 8, and 9 ~~and 8~~. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained. 1

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 7, 8, 9, and 10 ~~6, 7, and 8~~). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 11 ~~9~~) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System. 1 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

12

Power dependent MCPR limits ($MCPR_p$) are determined mainly by the one dimensional transient code (Ref. $\boxed{10}$). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level. 1

The MCPR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a ~~minimum~~ low recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor s provide~~s~~ rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required. 1 3

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

BASES

ACTIONS (continued)

B.1

If the MCPR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4, "Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter τ must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle.

MCPR
B 3.2.2

BASES

REFERENCES

1. NUREG-0562, June 1979.

E

2. NED~~O~~-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest ~~approved~~ version). 3

(revision specified in Specification 5.6.3) 4

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3. ~~F~~SAR, ~~Chapter~~ [4]. Section 3.2.4

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4. ~~F~~SAR, ~~Chapter~~ [6]. Section 6.2.6

2

U

5. ~~F~~SAR, Chapter [15]. 14

6. USAR, Chapter 14A. 1

INSERT 1

6. [Plant ~~specific single~~ loop operation].

INSERT 2

7. [Plant ~~specific load~~ line limit analysis].

INSERT 3

8. [Plant specific Average Power ~~Range~~ Monitor, Rod Block Monitor and Technical ~~Specification~~ Improvements (ARTS) Program].

2

11

9. NEDO-30130-A, "Steady State Nuclear Methods," May 1985. 1

12

10. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978. 1

2

INSERT 1

7. NEDE-23785-P (A), Revision 1, "The GESTR-LOCA and SAFER Models for Evaluation of the Loss-of-Coolant Accident (Volume III), SAFER/GESTR Application Methodology," October 1984.

2

INSERT 2

8. NEDC-30515, "GE BWR Extended Load Line Limit Analysis for Monticello Nuclear Generating Plant, Cycle 11," March 1984.

9. NEDC-31849P, including Supplement 1, "Maximum Extended Load Line Limit Analysis for Monticello Nuclear Generating Plant Cycle 15," June 1992.

2

INSERT 3

10. NEDC-30492-P, "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Monticello Nuclear Generating Plant," April 1984.

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.2 BASES, MINIMUM CRITICAL POWER RATIO (MCPR)

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases, which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.
2. The brackets are removed and the proper plant specific information/value is provided.
3. Typographical/grammatical error corrected.
4. Editorial change made for clarity.
5. Changes made to be consistent with changes made to the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.2, MINIMUM CRITICAL POWER RATIO (MCPR)

There are no specific NSHC discussions for this Specification.

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ATTACHMENT 3

ITS 3.2.3, LINEAR HEAT GENERATION RATE (LHGR)

Current Technical Specification (CTS) Markup
and Discussion of Changes (DOCs)

A.1

ITS | ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

If at any time during power operation, it is determined that the APLHGR limiting condition for operation is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two hours, reduce thermal power to less than 25% within the next four hours.

See ITS 3.2.1

A.2

B. Linear Heat Generation Rate (LHGR)

LCO 3.2.3 During power operation, the LHGR shall be less than or equal to the limits specified in the Core Operating Limits Report.

ACTION A If at any time during operation it is determined that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. If the LHGR is not returned to within the prescribed limits within 2 hours,

ACTION B reduce thermal power to less than 25% within the next 4 hours.

LA.1

A.3

B. Linear Heat Generation Rate (LHGR)

Add proposed SR 3.2.3.1 first Frequency

L.1

SR 3.2.3.1 The LHGR shall be checked daily during reactor operation at ≥25% of rated thermal power.

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DISCUSSION OF CHANGES
ITS 3.2.3, LINEAR HEAT GENERATION RATE (LHGR)

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.11.B states that the LHGR should not exceed limits during "power operation," which is defined in CTS 1.0.O as "above 1% rated thermal power." However, CTS 3.11.B only states to reduce THERMAL POWER to "less than 25%" if the limiting values for LHGR is being exceeded and the LHGRs are not returned to within limits within the specified time. ITS LCO 3.2.3 is applicable at THERMAL POWER ≥ 25% RTP. ITS 3.2.3 ACTION B requires a THERMAL POWER reduction to < 25% RTP if the LHGR(s) are not restored to within limits within the specified time limit of ACTION A. This changes the CTS by changing the Applicability from > 1% RATED THERMAL POWER ≥ 25% RTP.

The purpose of the CTS 3.11.B is to ensure the LHGRs are within limits when required. This changes the CTS by changing the Applicability from "power operation" to "≥ 25% RTP." This change is acceptable since at THERMAL POWER levels < 25% RTP the reactor is operating with substantial margin to the LHGR limits. For this reason there is no need to monitor LHGRs when THERMAL POWER < 25% RTP. This is also consistent with the Surveillance Frequency in CTS 4.11.B, which states to monitor LHGR at ≥ 25% RATED THERMAL POWER. This change simply aligns the Applicability with the CTS default action and Surveillance Frequency. Therefore, this change is considered a presentation preference change only and, as such, is considered an administrative change.

A.3 CTS 3.11.B states "Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits." ITS 3.2.3 does not include this statement. This changes the CTS by deleting this statement.

The purpose of this CTS 3.11.B statement is to identify the importance of monitoring the LGHRs to verify they are restored to prescribed limits. After they are within limits, it is obvious that the action can be exited. ITS LCO 3.0.1 and LCO 3.0.2 have been added to the Technical Specifications as indicated in the Discussion of Changes for ITS Section 3.0. ITS LCO 3.0.1 states " LCOs shall be met during the MODES or other specified conditions in the Applicability," and LCO 3.0.2 states "Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met" and "If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not require." The CTS 3.11.B guidance is provided in the ITS generic guidelines of LCO 3.0.1 and LCO 3.0.2. In addition, the only way to confirm the LHGRs have been restored to within limits is to perform a Surveillance; thus, it is not necessary to be specifically stated.

DISCUSSION OF CHANGES
ITS 3.2.3, LINEAR HEAT GENERATION RATE (LHGR)

Therefore, this change is considered a presentation preference change only and, as such, is considered an administrative change.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 3.11.B states that if at any time during power operation it is determined that the limiting value for LHGR limiting condition for operation is being exceeded, "action shall be initiated within 15 minutes to restore operation to within the prescribed limits." ITS 3.2.3 does not include this 15 minute action. This changes the CTS by relocating the procedural detail that "action shall be initiated within 15 minutes to restore operation to within the prescribed limits" to the Bases in the form of a discussion that "prompt action should be taken to restore the LHGR(s) to within the required limits."

The removal of this detail for performing actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to restore the LHGRs to within limits in 2 hours, consistent with the CTS actions. Also, this change is acceptable because this type of procedural detail will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because a procedural detail for meeting Technical Specification requirements is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 4.11.B requires the LHGR to be determined daily during reactor operation at ≥ 25% rated thermal power. ITS SR 3.2.3.1 requires the same verification "once within 12 hours after ≥ 25% RTP and 24 hours thereafter." This changes the CTS by allowing the reactor to reach and exceed a THERMAL POWER level of 25% RTP without completing the Surveillance.

The purpose of CTS 4.11.B is to ensure all LHGRs are within limits before THERMAL POWER is ≥ 25% RTP. This change is acceptable because the new

DISCUSSION OF CHANGES
ITS 3.2.3, LINEAR HEAT GENERATION RATE (LHGR)

Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of fuel reliability. This change allows the plant to increase THERMAL POWER ≥ 25% RTP without completing the Surveillance. However, after 25% RTP is achieved the verification must be performed within 12 hours and every 24 hours thereafter. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

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Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR) (Optional) 1

3.11.B LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

3.11.B APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.11.B	A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	2 hours
3.11.B	B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY
4.11.B	SR 3.2.3.1	Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.3, LINEAR HEAT GENERATION RATE (LHGR)

1. This reviewer's type of note has been deleted. This is not meant to be retained in the final version of the plant specific submittal.

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Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR) ~~(Optional)~~ 1

BASES

BACKGROUND

node The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the normal operations and anticipated operating conditions identified in Reference 1. 5 3

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1 and 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; ~~and~~ 6

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of [1%] plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3). 2

Fuel design evaluations have been performed and demonstrate that the [1%] fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs, plus an allowance for densification power spiking. 2

The LHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.

APPLICABILITY

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 25% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 25% RTP.

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limits such that the plant is operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LHGR out of specification.

B.1

If the LHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER TO < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

s are They are

The LHGR ~~is~~ required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. ~~It~~ ~~is~~ compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slow changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels. 4

BASES

REFERENCES

3 U

1. ~~F~~SAR, ~~Section [ ]~~. Chapter 14 2

2. ~~F~~SAR, ~~Section [ ]~~. Chapter 3 2

3. NUREG-0800, Section II.A.2(g), Revision 2, July 1981.

JUSTIFICATION FOR DEVIATIONS
ITS 3.2.3 BASES, LINEAR HEAT GENERATION RATE (LHGR)

1. This reviewer's type of note has been deleted. This is not meant to be retained in the final version of the plant specific submittal.
2. The brackets are removed and the proper plant specific information/value is provided.
3. Changes are made (additions, deletions, and/or changes) to the ISTS Bases, which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.
4. Change made to be consistent with the Specification.
5. Editorial change made for clarity.
6. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.2.3, LINEAR HEAT GENERATION RATE (LHGR)

There are no specific NSHC discussions for this Specification.

ATTACHMENT 4

Improved Standard Technical Specifications (ISTS) not adopted in the Monticello ITS

ISTS 3.2.4, Average Power Range Monitor (APRM) Gain and Setpoints

ISTS 3.2.4 Markup and Justification for Deviations (JFDs)

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints (Optional)

LCO 3.2.4

a. MFLPD shall be less than or equal to Fraction of RTP, or

b. Each required APRM setpoint specified in the COLR shall be made applicable, or

c. Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times MFLPD.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 25% RTP.	4 hours

1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	-----NOTE----- Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4 Item b or c requirements. ----- Verify MFLPD is within limits.	Once within 12 hours after ≥ 25% RTP **AND** 24 hours thereafter
SR 3.2.4.2	-----NOTE----- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4 Item a requirements. ----- Verify APRM setpoints or gains are adjusted for the calculated MFLPD.	12 hours

1

JUSTIFICATION FOR DEVIATIONS
ISTS 3.2.4, AVERAGE POWER RANGE MONITOR (APRM) GAIN AND SETPOINTS

1. ISTS 3.2.4 has not been adopted since it is not applicable to Monticello. The requirements for Average Power Range Monitor (APRM) Gain and Setpoints have been previously deleted from the Monticello Technical Specifications as a result of License Amendment 29, dated November 16, 1984.

ISTS 3.2.4 Bases Markup and Justification for Deviations (JFDs)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints (Optional)

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design," GDC 13, "Instrumentation and Control," GDC 20, "Protection System Functions," and GDC 23, "Protection against Anticipated Operation Occurrences" (Ref. 1). This LCO is provided to require the APRM gain or APRM flow biased scram setpoints to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP), where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when: 1

$$\frac{\text{MFLPD}}{\text{FRTP}} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by a gain adjustment on the APRMs or adjustment of the APRM setpoints. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP and thus maintains RTP margins for APLHGR and MCPR.

The normally selected APRM setpoints position the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The setpoints are flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow, the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM setpoints are supported by the analyses presented in References 1 and 2 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR and MCPR), at

BASES

BACKGROUND (continued)

rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the flow biased APRM scram setpoints may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

The APRM neutron flux signal is also adjusted to more closely follow the fuel cladding heat flux during power transients. The APRM neutron flux signal is a measure of the core thermal power during steady state operation. During power transients, the APRM signal leads the actual core thermal power response because of the fuel thermal time constant. Therefore, on power increase transients, the APRM signal provides a conservatively high measure of core thermal power. By passing the APRM signal through an electronic filter with a time constant less than, but approximately equal to, that of the fuel thermal time constant, an APRM transient response that more closely follows actual fuel cladding heat flux is obtained, while a conservative margin is maintained. The delayed response of the filtered APRM signal allows the flow biased APRM scram levels to be positioned closer to the upper bound of the normal power and flow range, without unnecessarily causing reactor scrams during short duration neutron flux spikes. These spikes can be caused by insignificant transients such as performance of main steam line valve surveillances or momentary flow increases of only several percent. 1

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR and MCPR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

FSAR safety analyses (Refs. 2 and 3) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR and MCPR) occurs. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of either the APLHGR or the MCPR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be

BASES

APPLICABLE SAFETY ANALYSES (continued)

hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the SLs could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the FRTP, or the flow biased APRM scram level is required to be reduced by the ratio of FRTP to the core limiting MFLPD. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the flow biased APRM scram setpoints, dependent on the increased peaking that may be encountered.

The APRM gain and setpoints satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Meeting any one of the following conditions ensures acceptable operating margins for events described above: 1

a. Limiting excess power peaking,

b. Reducing the APRM flow biased neutron flux upscale scram setpoints by multiplying the APRM setpoints by the ratio of FRTP and the core limiting value of MFLPD, or

c. Increasing APRM gains to cause the APRM to read greater than 100 times MFLPD (in %). This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM flow biased scram setpoints are reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM flow biased scram setpoints. Adjusting APRM gain or setpoints is equivalent to MFLPD less than or equal to FRTP, as stated in the LCO.

BASES

LCO (continued)

For compliance with LCO Item b (APRM setpoint adjustment) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are required to be adjusted. In addition, each APRM may be allowed to have its gain or setpoints adjusted independently of other APRMs that are having their gain or setpoints adjusted.

APPLICABILITY

The MFLPD limit, APRM gain adjustment, and APRM flow biased scram and associated setdowns are provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the reactor is operating at ≥ 25% RTP.

ACTIONS

A.1

If the APRM gain or setpoints are not within limits while the MFLPD has exceeded FRTP, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit may be reduced. Therefore, prompt action should be taken to restore the MFLPD to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses. 1

The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or setpoints to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

B.1

If MFLPD cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

BASES

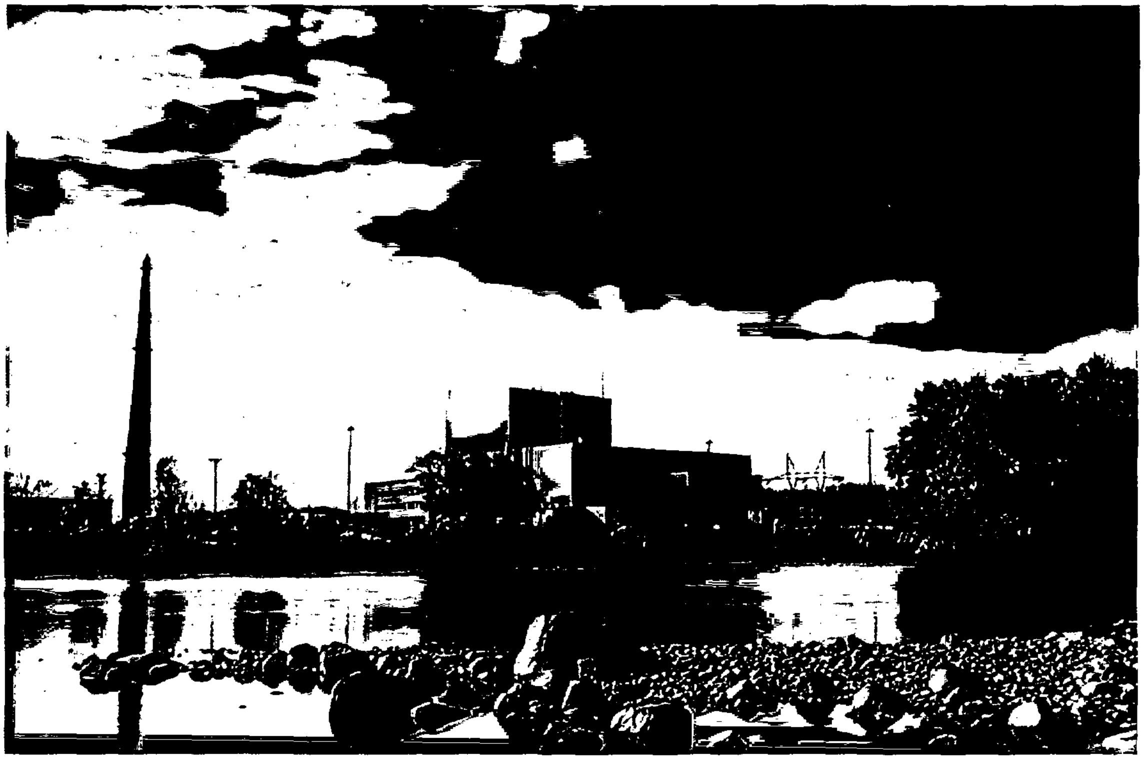
SURVEILLANCE REQUIREMENTS	SR 3.2.4.1 and SR 3.2.4.2 The MFLPD is required to be calculated and compared to FRTP or APRM gain or setpoints to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are only required to determine the MFLPD and, assuming MFLPD is greater than FRTP, the appropriate gain or setpoint, and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram circuitry. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels. The 12 hour Frequency of SR 3.2.4.2 requires a more frequent verification than if MFLPD is less than or equal to fraction of rated power (FRP). When MFLPD is greater than FRP, more rapid changes in power distribution are typically expected.
REFERENCES	1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 23. 2. FSAR, Section [ ]. 3. FSAR, Section [ ].

1

JUSTIFICATION FOR DEVIATIONS
ISTS 3.2.4 BASES, AVERAGE POWER RANGE MONITOR (APRM) GAIN AND SETPOINTS

1. Changes are made to be consistent with changes made to the Specification.

IMPROVED TECHNICAL SPECIFICATIONS



MONTICELLO NUCLEAR GENERATING PLANT

VOLUME 8
BOOK 1
REVISION 1

ITS Section 3.3, - Instrumentation

NMC
Committed to Nuclear Excellence

Summary of Changes
ITS Section 3.3

Change Description	Affected Pages
The changes described in the NMC response to Question 200510281240 have been made. The Degraded Voltage Allowable Values have been changed to be consistent with the most recent setpoint calculation.	Pages 686 and 700 of 763
The changes described in the NMC response to Question 200510281246 have been made. The ADS Core Spray and RHR Pumps High Discharge Pressure Allowable Values have been changed to be consistent with the most recent setpoint calculations and Notes have been added to the CHANNEL CALIBRATION Surveillance, similar to those Notes in draft TSTF-493 Rev. 0.	Pages 321, 324, 335, 336, 354, 356, 357, 358, 359, 360, 363, 411, 412, and 414 of 763
The changes described in the NMC response to Question 200510281248 have been made. Notes have been added to the CHANNEL CALIBRATION Surveillance for the Core Spray and RHR High Reactor Steam Dome Pump and Valve Permissives Functions, similar to those Notes in draft TSTF-493 Rev. 0.	Pages 324, 341, 342, 350, 351, 352, 353, 354, 356, 357, 359, 363, 411, 412, and 414 of 763
The changes described in the NMC response to Question 200601061257 have been made. ITS SR 3.3.1.2.5 has been modified to only allow the determination of signal to noise ratio portion of the SR to not be met with less than or equal to two fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant.	Page 128 of 763
The changes described in the NMC response to Question 200601201447 have been made. Minor editorial changes are made. Subsequent to the NMC response, the following additional editorial changes to the ITS Bases Markups were identified: (a) ITS 3.3.1.1 page B 3.3.1.1-3, last paragraph - logic channels "A.3" and "B.3" should be "A3" and "B3"; (b) ITS 3.3.1.1 Insert Page B 3.3.1.1-9, fourth sentence - "of Reference (Ref. 9)" should be "of Reference 9"; (c) ITS 3.3.1.1 page B 3.3.1.1-14, fourth paragraph - "Reactor Vessel Water - Low Low" should be "Reactor Vessel Water Level - Low Low"; (d) ITS 3.3.1.1 page B 3.3.1-1-29, third paragraph - "an 24 month" should be "a 24 month"; (e) ITS 3.3.3.2 Insert Page B 3.3.3.2-6, INSERT 4, item c - for clarity, the parenthetical phrase "(includes RHR Service Water controls)" has been added; (f) ITS 3.3.4.1 Insert Page B 3.3.4.2-2 - the first sentence is duplicative of the second sentence and has been deleted; (g) ITS 3.3.5.1 Insert Page B 3.3.5.1-10a, third paragraph of INSERT 8 - "this optimizes the" should be "thus optimizing the"; (h) ITS 3.3.6.3 Insert Page B 3.3.6.3-3b, first paragraph, second sentence - "occurs if a there is" should be "occurs if there is"; (i) ITS 3.3.8.1 page B 3.3.8.1-2, last paragraph - inserted words "4.16 Essential Bus" should be "4.16 kV Essential Bus"; and (j) ITS 3.3.8.2 page B 3.3.8.2-2, LCO Section, second paragraph, inserted words "or SR 3.3.8.2.3" should be "and SR 3.3.8.2.3".	Pages 13, 40, 56, 60, 71, 77, 88, 90, 99, 170, 210, 238, 242, 243, 248, 266, 267, 271, 272, 286, 299, 302, 330, 345, 352, 354, 362, 363, 370, 371, 378, 379, 382, 383, 387, 388, 389, 392, 405, 406, 408, 492, 495, 497, 503, 505, 512, 528, 530, 535, 538, 603, 631, 632, 656, 703, 705, 707, 726, 727, 729, 733, 735, and 738 of 763

Summary of Changes
ITS Section 3.3

Change Description	Affected Pages
The changes described in the NMC response to Question 200603161318 have been made. Various A DOCs and L DOCs have been changed to clarify the change from Trip Settings to Allowable Values as the OPERABILITY value. Note: ITS 3.3.1.1 DOC A.16 change has not been made since it is superseded by the NMC response to 200604171314.	Pages 40, 41, 160, 167, 168, 285, 290, 291, 332, 428, 477, 488, 609, 693, and 720 of 763
The change described in the NMC response to Question 200604171314 has been made. ITS 3.3.1.1 DOC A.16 has been modified for clarity.	Page 21 of 763

ATTACHMENT 1

VOLUME 8

MONTICELLO
IMPROVED TECHNICAL
SPECIFICATIONS CONVERSION

ITS SECTION 3.3
INSTRUMENTATION

Revision 1

LIST OF ATTACHMENTS

1. ITS 3.3.1.1
2. ITS 3.3.1.2
3. ITS 3.3.2.1
4. ITS 3.3.2.2
5. ITS 3.3.3.1
6. ITS 3.3.3.2
7. ITS 3.3.4.1
8. ITS 3.3.5.1
9. ITS 3.3.5.2
10. ITS 3.3.6.1
11. ITS 3.3.6.2
12. ITS 3.3.6.3
13. ITS 3.3.7.1
14. ITS 3.3.8.1
15. ITS 3.3.8.2
16. Improved Standard Technical Specifications (ISTS) not adopted in the Monticello ITS

ATTACHMENT 1

ITS 3.3.1.1, Reactor Protection System (RPS) Instrumenation

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3.3.1.1 **3.1 REACTOR PROTECTION SYSTEM**	**4.1 REACTOR PROTECTION SYSTEM**
Applicability: Applies to the instrumentation and associated devices which initiate a reactor scram.	Applicability: Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.
Objective: To assure the operability of the reactor protection system.	Objective: To specify the type and frequency of surveillance to be applied to the instrumentation that initiates a scram to verify its operability.
Specification:	Specification:
LCO 3.3.1.1 — A. The ~~setpoints~~, minimum number of trip systems and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1. (APPLICABILTY) The time from initiation of any channel trip to the de-energization of the scram pilot valve solenoids shall not exceed 50 milliseconds. (SR 3.3.1.1.14)	A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively. (Note 1 to Surveillance Requirements)

A.16

Allowable Values

within limits

LA.2

A.2

See ITS Chapter 1.0

LA.1

Add proposed SR 3.3.1.1.12 — M.1

L.1

A.1

ITS

LA.2

A.3

ITS	3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
ACTION NOTE	B. Upon discovery that the requirements for the number of operable or operating trip systems or instrument channels are not satisfied, action shall be initiated as follows:	B. (DELETED)
ACTION A	1. With one required instrument channel inoperable in one or more trip functions, place the inoperable channel(s) or trip system in the tripped condition within 12 hours, or	
ACTIONS B and C	2. ~~With more than one instrument channel inoperable for one or more trip functions, immediately satisfy the minimum requirements by placing appropriate channel(s), or trip system(s) in the tripped condition, or~~	
ACTIONS D, E, F, G, and H	3. Place and maintain the plant under the specified required conditions ~~using normal operating procedures.~~	Add proposed Completion Times for Required Actions E.1, F.1, G.1, and H.1

M.2

L.2

M.3

Attachment 1, Volume 8, Rev. 1, Page 6 of 763

A.1

ITS

A.16

Allowable Value

APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS

A.2

LA.2

Table 3.3.1.1-1

CONDITIONS REFERENCED FROM REQUIRED ACTION D.1

A.1

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

ITS	Trip Function	~~Limiting Trip Settings~~	Modes in which function must be Operable or Operating** Refuel (3)	Startup	Run	Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1)	~~Required Condition*~~
10	1. Mode Switch in Shutdown		X	X	~~X~~	1	1	A
11	2. Manual Scram		X	X	~~X~~	1	1	A
1	3. Neutron Flux IRM ~~(See Note 2)~~ LA.4	≤ ~~120~~/125 of full scale [122] L.12	X [2, 5[(a)] L.3]	X		4	3 LA.2	A
1.a	a. High-High	~~AND < 20% of Rated Thermal Power~~ LA.3						
1.b	b. Inoperative							
2	4. Flow Referenced Neutron Flux APRM ~~(See Note 5)~~ LA.4	≤ [0.66W+~~65.6~~] [67.6] L.12 %Rated Thermal Power for two loop operation OR			~~X~~ MODE 1	3	2	~~A or~~ B A.15
2.a	a. High-High							
2.b	b. Inoperative							
	Note (b)	≤ [0.66(W-5.4)+~~65.6~~] %Rated Thermal Power for single loop operation						
	LA.3	~~Where: W=percent of recirculation drive flow to produce a core flow of 57.6x10[6] lbm/hr~~	MODE 2	M.5	MODE 1			
2.a	c. High Flow Clamp	≤ ~~120~~ % [122]						
3	5. High Reactor Pressure ~~(See Note 9)~~ M.5	≤ 1075 psig L.12	X	X(i) L.4	~~X~~(i)	2	2	A

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ITS

A.1 A.16 A.2 LA.2 A.1

Allowable Value

APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS

CONDITIONS REFERENCED FROM REQUIRED ACTION D.1

Table 3.3.1.1-1

TABLE 3.1.1 - CONTINUED

ITS	Trip Function	~~Limiting Trip Settings~~	Modes in which function must be Operable or Operating** Refuel (3)	Startup	Run	Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1)	~~Required Condition*~~
6	6. High Drywell Pressure ~~(See Note 4)~~	≤ 2 psig	X	X(a)(f)	X(a)(f)	2	2	A
4	7. Reactor Low Water Level	≥ 7 in.	X	X(f)	X(f)	2	2	A
7	8. Scram Discharge Volume High Level							
	a. East	≤ 56 gal. (8)	X(a)	X(f)	X(f)	2	2	A
	b. West	≤ 56 gal. (8)	X(a)	X(f)	X(f)	2	2	A
	~~9. Turbine Condenser Low Vacuum~~	≥ 22 in. Hg	X(b)	X(b,f)	X(f)	2	2	A or C
5	10. Main Steamline Isolation Valve Closure	≤ 10% Valve Closure	X(b)	X(b)	X	8	8	~~A or~~ C
9	11. Turbine Control Valve Fast Closure	(See Note 7)			X(c,f)	2	2	D
8	12. Turbine Stop Valve Closure	≤ 10% Valve Closure			X(d)	4	4	D

M.6 M.7 M.8 2 2 LA.5 L.3 2 5(e) L.4 MODE 1 L.4 LA.2 A.4 R.1 A.15 2(c) MODE 1 A.5 L.4 LA.2 A.12 > 45% RTP M.9 A.14

Add proposed Allowable Value

NOTES:

Surveillance Requirements Note 1

1. ~~There shall be two operable or tripped trip systems for each function.~~ A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided that at least one other operable channel in the same trip system is monitoring that parameter.
2. ~~For an IRM channel to be considered operable, its detector shall be fully inserted.~~
3. In the refueling mode with the reactor subcritical and reactor water temperature less than 212°F, only the following trip functions need to be operable: (a) Mode Switch in Shutdown, (b) Manual Scram, (c) High Flux IRM, (d) Scram Discharge Volume High Level
4. ~~Not required to be operable when primary containment integrity is not required.~~
5. ~~To be considered operable, an APRM must have at least 2 LPRM inputs per level and at least a total of 14 LPRM inputs, except that channels 1, 2, 5, and 6 may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.~~

LA.2 A.6 LA.4 L.3 M.6 LA.4

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A.1

ITS

Table 3.1.1 - Continued

6. Deleted. M.9

Table 3.3.1.1-1 Function 9 — 7. Trips upon loss of oil pressure to the acceleration relay.

8. ~~Limited trip setting refers to the volume of water in the discharge volume receiver tank and does not include the volume in the lines to the level switches.~~ LA.5

9. ~~High reactor pressure is not required to be operable when the reactor vessel head is unbolted.~~ M.5

* Required Conditions when minimum conditions for operation are not satisfied. (ref. 3.1.B) A.13 A.8

ACTIONS G and H — A. All operable control rods fully inserted. in 12 hours for Action G and Immediately for Action H M.3

ACTION F — B. Power on IRM range or below and reactor in Startup, Refuel, or Shutdown mode in 6 hours A.15

ACTION F — C. Reactor in Startup or Refuel mode and pressure below 600 psig. in 12 hours M.3

ACTION E — D. Reactor power less than 45% ~~(798.75 MWt.)~~. in 4 hours A.7

** Allowable Bypass Conditions

It is permissible to bypass:

a. ~~The scram discharge volume High Water Level scram function in the refuel mode to allow reactor protection system reset. A rod block shall be applied while the bypass is in effect.~~ L.3

Table 3.3.1.1-1 Footnote (c) — b. The ~~Low Condenser vacuum~~ and MSIV closure scram function in the Refuel and Startup modes if reactor pressure is below 600 psig. R.1

c. Deleted.

Table 3.3.1.1-1 Functions 8 and 9 — d. The turbine stop valve closure and fast control valve closure scram functions when the reactor thermal power is ≤ 45% ~~(798.75 MWt)~~ A.7

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A.1

Table 3.1.1 - Continued

e. ~~The high drywell pressure scram functions in the Startup and Run modes when necessary during purging for containment inerting or de-inerting only by closing the manual containment isolation valves. Verification of the bypass condition shall be noted in the control room log.~~

M.8

f. ~~One instrument channel for the functions indicated in the table to allow completion of surveillance testing, provided that:~~

1. ~~Redundant instrument channels in the same trip system are capable of initiating the automatic function and are demonstrated to be operable either immediately prior or immediately subsequent to applying the bypass.~~

2. ~~While the bypass is applied, surveillance testing shall proceed on a continuous basis and the remaining instrument channels initiating the same function are tested prior to any other. Upon completion of surveillance testing, the bypass is removed.~~

L.4

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A.1

ITS

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Table 3.3.1.1-1

TABLE 4.1.1

SCRAM INSTRUMENT FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENTATION AND CONTROL CIRCUITS

	INSTRUMENTATION CHANNEL	FUNCTIONAL TEST	MINIMUM FREQUENCY (4)
3	High Reactor Pressure	Trip Channel and Alarm	Quarterly -SR 3.3.1.1.7
6	High Drywell Pressure	Trip Channel and Alarm	Quarterly -SR 3.3.1.1.7
4	Low Reactor Water Level (2, 5)	Trip Channel and Alarm	Quarterly -SR 3.3.1.1.7
7.a, 7.b	High Water Level in Scram Discharge Volume	Trip Channel and Alarm	Quarterly -SR 3.3.1.1.7
	~~Condenser Low Vac~~	~~Trip Channel and Alarm~~	~~Once each month~~
5	Main Steam Line Isolation Valve Closure	Trip Channel and Alarm	Quarterly -SR 3.3.1.1.7
8	Turbine Stop Valve Closure	Trip Channel and Alarm	Quarterly -SR 3.3.1.1.7
11	Manual Scram	Trip Channel and Alarm	~~Weekly~~ 31 days -SR 3.3.1.1.5
9	Turbine Control Valve Fast Closure	Trip Channel and Alarm	Quarterly -SR 3.3.1.1.7
2.a, 2.b	APRM/Flow Reference (5)	Trip Output Relays	Quarterly -SR 3.3.1.1.7
1.a, 1.b	IRM (5)	Trip Channel and Alarm	Note 3 -SR 3.3.1.1.3
10	Mode Switch in Shutdown	Place mode switch in shutdown	Every ~~Operating Cycle~~ 24 months -SR 3.3.1.1.10

L.5

A.10

A.11

R.1

L.5

L.14

A.11

A.9

Add proposed SR 3.3.1.1.13 for Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure - Low Functions

M.11

Add proposed SR 3.3.1.1.4 for all automatic Functions

M.10

Add proposed Note to SR 3.3.1.1.3 M.13

TABLE 4.1.1 (Continued)

Add proposed SR 3.3.1.1.1 for Functions 1.a and 2.a M.15

Note 1: ~~Deleted.~~

7 days

SR 3.3.1.1.1 Note 2: A sensor check shall be performed on low reactor water level once per ~~day~~. 12 hours M.12

SR 3.3.1.1.3 Note 3: Perform functional test prior to every startup ~~and demonstrate that the IRM and APRM channels overlap at least 1/2 decade prior to every normal shutdown.~~ L.6

~~Note 4: Functional tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.~~ A.10

~~Note 5: A functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response, alarm, and/or initiating action.~~ A.11

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A.1

ITS

Table 3.3.1.1-1

M.14

TABLE 4.1.2

SCRAM INSTRUMENT CALIBRATION

MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

A.10

	INSTRUMENT CHANNEL	GROUP	CALIBRATION METHOD	MINIMUM FREQUENCY (2)	
2.a, 2.b	APRM ← Add proposed SR 3.3.1.1.6 and SR 3.3.1.1.9	B	Heat Balance	Once every 3 days (4) -SR 3.3.1.1.2	7 days — L.7; LA.7
1.a, 1.b	IRM	B	Heat Balance	See Note 1 -SR 3.3.1.1.11	LA.6
3	High Reactor Pressure	A	Pressure Standard	Every 3 months -SR 3.3.1.1.9	
6	High Drywell Pressure	A	Pressure Standard	Every 3 months -SR 3.3.1.1.9	L.9
4	Low Reactor Water	B	Pressure Standard	Every Operating Cycle -SR 3.3.1.1.11 Transmitter Every 3 months - Trip Unit -SR 3.3.1.1.8	24 months — A.9; L.9
7.a, 7.b	High Water Level in Scram Discharge	A or B	Water Level	Every 3 months -SR 3.3.1.1.9	
	~~Condenser Low Vacuum~~	~~A~~	~~Vacuum Standard~~	~~Every 3 months~~	R.1
5	Main Steamline Isolation Valve Closure	A	Observation	Every Operating Cycle -SR 3.3.1.1.11	24 months — A.9
9	Turbine Control Valve Fast Closure	A	Pressure Standard	Every 3 months -SR 3.3.1.1.9	LA.6
8	Turbine Stop Valve Closure	A	Observation	Every Operating Cycle -SR 3.3.1.1.11	
2.a	Recirculation Flow Meters & Flow Instrumentation	-	Pressure Standard	Every 3 months -SR 3.3.1.1.9	L.9

Add proposed Note 1 to SR 3.3.1.1.11 — L.13

Add proposed Note 2 to SR 3.3.1.1.11

Notes:

SR 3.3.1.1.11 1. Perform calibration test during every startup and normal shutdown. ← 24 months — L.10

2. ~~Calibration tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.~~ — A.10

3. (Deleted).

Add proposed Note to SR 3.3.1.1.2 — L.8

SR 3.3.1.1.2 4. ~~This calibration is performed by taking~~ a heat balance ~~and adjusting the APRM to agree with the heat balance.~~ ~~Alarms and trips will be verified and calibrated if necessary during functional testing.~~ — LA.7; L.5

Add proposed 2% RTP acceptance criteria — M.4

~~*GROUPS:~~

~~A. Passive type devices.~~

~~B. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.~~ — LA.6

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A.1

ITS

See ITS Chapter 1.0

Add proposed Note 2 to SR 3.3.1.1.14

SR 3.3.1.1.14

F. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, accuracy, and response time to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm or trip. Response time is not part of the routine instrument calibration but will be checked once per cycle.

24 months

on a STAGGERED TEST BASIS

L.11

A.9

G. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

H. -Deleted-

I. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation, the safety limits will never be exceeded.

J. Minimum Critical Power Ratio (MCPR) - The minimum critical power ratio is the value of critical power ratio associated with the most limiting assembly in the reactor core. Critical power ratio (CPR) is the ratio of that power in a fuel assembly which is calculated by the GEXL correlation to cause some point in the assembly to experience boiling transition to the actual assembly operating power.

K. Mode - The reactor mode is that which is established by the mode-selector switch.

L. Operable - A system, subsystem, train, component or device shall be Operable or have Operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

See ITS Chapter 1.0

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DISCUSSION OF CHANGES
ITS 3.3.1.1, REACTOR PROTECTION SYSTEM (RPS) INSTRUMENTATION

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.1.A specifies the applicability requirements for the RPS Instrumentation Functions based on "each position of the reactor mode switch as indicated in Table 3.1.1." ITS Table 3.3.1.1-1 either specifies the Applicable MODES as defined in ITS Section 1.1 or other specified conditions. This changes the CTS by using the defined term of MODES in the Applicability, whenever possible. Changes to the actual requirements for when the RPS instrumentation must be OPERABLE are discussed below in other DOCs.

This change is considered acceptable because the Applicability requirements for the RPS Instrumentation Functions in ITS 3.3.1.1 adopts the use of the new defined terms in ITS Section 1.1 (i.e., MODES). Any technical changes to the Applicability of the RPS Instrumentation Functions are discussed below. This change is designated as administrative change and is acceptable because it does not result in technical changes to the CTS.

A.3 CTS 3.1.B states "Upon discovery that the requirements for the number of operable or operating trip systems or instrument channels are not satisfied, action shall be initiated as follows:" ITS 3.3.1.1 ACTIONS Note states "Separate Condition entry is allowed for each channel." This changes the CTS by clarifying that separate Condition entry is allowed for each channel.

The purpose of CTS 3.1.B is to specify which actions are to be taken when a channel becomes inoperable. This proposed change to the CTS 3.1.B Actions provides more explicit instructions for proper application of the Actions for Technical Specification compliance. In conjunction with the proposed Specification 1.3, "Completion Times," the ITS 3.3.1.1 ACTIONS Note ("Separate Condition entry is allowed for each....."), the wording for ACTION A ("One or more required channels"), and ITS 3.3.1.1 ACTIONS B and C ("One or more Functions with one or more required channels") provide direction consistent with the intent of the existing Actions for an inoperable RPS instrumentation channel. It is intended that each inoperable channel is allowed a certain time to complete the Required Actions. Since this change only provides more explicit direction of the current interpretation of the existing specifications, this change is considered administrative. Any change to the actual time to perform the CTS 3.1.B actions is discussed in other DOCs. This change is administrative because it does not result in technical changes to the CTS.

A.4 CTS Table 3.1.1 Trip Function 8.a requires two "East" Scram Discharge Volume High Level channels to be OPERABLE in each trip system while CTS Table 3.1.1 Trip Function 8.b requires two "West" Scram Discharge Volume High Level

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channels to be OPERABLE in each trip system. ITS Table 3.3.1.1-1 Function 7.a requires two Resistance Temperature Detector channels to be OPERABLE in each trip system while Function 7.b requires two Float Switch channels to be OPERABLE in each trip system. This changes the CTS by specifying the "type of channels" instead of the "location" of the channels.

The purpose of the CTS Table 3.1.1 Trip Function 8 channel requirements are to ensure that no single instrument failure will preclude a scram from these Functions on a valid signal. This change is acceptable since the number of required channels will be the same. The design includes a "West" and "East" Scram Discharge Volume. Each discharge volume is instrumented with two resistance temperature detectors and two float switch detectors (one per trip system for each type). Each RPS trip system will include input from each type of detector from both discharge volumes. Therefore, each trip system will have a total of 4 channels. Since, the ITS also requires a total of four detectors in each trip system, this change is simply an administrative change to be consistent with the ISTS format. This change is administrative because it does not result in technical changes to the CTS.

A.5 CTS Table 3.1.1 requires the Main Steamline Isolation Valve Closure Trip Function (Trip Function 10) to be OPERABLE when the reactor mode switch is in the Refuel, Startup, and Run positions. However, CTS Table 3.1.1 Note (b) states that the MSIV closure scram function may be bypassed in the Refuel and Startup modes if reactor pressure is below 600 psig. Furthermore, CTS Table 3.1.1 Note (3) states that the only RPS Trip Functions that are required to be OPERABLE when in the refueling mode with the reactor subcritical and reactor water temperature less than 212°F are Mode Switch in Shutdown, Manual Scram, High Flux IRM (i.e., Neutron Flux IRM High - High and Neutron Flux IRM Inoperative), and Scram Discharge Volume High Level. ITS Table 3.3.1.1-1 Function 5 requires the Main Steam Isolation Valve - Closure Function to be OPERABLE in MODE 1 and MODE 2 with reactor pressure $\geq$ 600 psig (as stated in ITS Table 3.3.1.1-1 Note c). This changes the CTS by clearly stating the Applicability of the Main Steamline Isolation Valve Closure Trip Function.

The purpose of CTS Table 3.1.1 Trip Function 10 is to provide an anticipatory scram since the normal heat sink will be lost when the MSIVs close. As stated in CTS Table Note (b), the Function is only required when reactor pressure is $\geq$ 600 psig. In the ITS, MODE 2 is defined as the reactor mode switch in Startup/Hot Standby or Refuel (with all head closure bolts tensioned). Therefore, ITS MODE 2 covers the CTS conditions of the reactor mode switch in the Refuel or Startup positions with reactor pressure $\geq$ 600 psig. Therefore this change is acceptable and is simply an administrative change to be consistent with the ISTS format. This change is designated as administrative because it does not result in a technical change to the CTS.

A.6 CTS Table 3.1.1 Note (1) states that a channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided "that at least one other operable channel in the same trip system is monitoring that parameter." ITS 3.3.1.1 Surveillance Requirements Table Note 2 states that when a channel is placed in an

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inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided "the associated Function maintains RPS trip capability." This changes the CTS by replacing the words "at least one other operable channel in the same trip system is monitoring that parameter" with "the associated Function maintains RPS trip capability."

The purpose of CTS Table 3.1.1 Note (1) is to allow 6 hours to perform Surveillance testing without entering the actions. Most of the RPS Functions contain at least four channels arranged in a one-out-of-two taken twice logic configuration. Therefore, the design ensures RPS trip capability is maintained when one channel is placed in an inoperable condition. While some Trip Functions include more than two channels in a trip system, the intent of the CTS Note is to ensure RPS trip capability is not lost. The proposed wording includes the same restrictions that are in the CTS. This change is acceptable since the proposed wording is consistent with the current requirements. This change is designated as administrative because it does not result in technical changes to the CTS.

A.7 CTS Table 3.1.1 requires the Turbine Control Valve Fast Closure and Turbine Stop Valve Closure Trip Functions (Trip Functions 11 and 12) to be OPERABLE when the reactor mode switch is in the Run position. However, CTS Table 3.1.1 footnote **.d states that these scram functions may be bypassed when the reactor thermal power is ≤ 45%. The Note also provides a parenthetical reference that 45% rated thermal power is equivalent to 798.75 MWt. CTS Table 3.1.1 Required Condition D also provides a similar parenthetical reference. ITS Table 3.3.1.1-1 Functions 8 and 9 specify the Applicability to be > 45% RTP and ITS 3.3.1.1 ACTION E requires the unit to be ≤ 45% RTP. This changes the CTS by deleting the actual thermal power level (798.75 MWt) from the Applicability and Action.

"Rated Thermal Power" is a definition in the CTS, and contains the actual value for "Rated Thermal Power" in MWt. This change is acceptable because this definition and the RATED THERMAL POWER value in MWt are retained in the ITS 1.1. The actual thermal power level is not necessary since the value can be easily calculated. This change is designated as administrative because it does not result in technical changes to the CTS.

A.8 When the requirements of CTS 3.1.B are not met for the Mode Switch in Shutdown, Manual Scram, Neutron Flux IRM High - High, Neutron Flux IRM Inoperable, High Reactor Pressure, High Drywell Pressure, Reactor Low Water Level, and Scram Discharge Volume High Level (East and West) Trip Functions (CTS Table 3.1.1 Trip Functions 1, 2, 3.a, 3.b, 5, 6, 7, 8.a, and 8.b), CTS Table 3.1.1 (Required Condition A) requires all OPERABLE control rods to be fully inserted. Under similar conditions in the ITS (i.e., the Required Actions and associated Completion Times of ACTIONS A, B, and C are not met) and when the unit is in MODE 1 or 2, ITS 3.3.1.1 ACTION G will require the unit to be in MODE 3. This changes the CTS by specifying the unit must be in MODE 3 instead of all OPERABLE control rods must be fully inserted.

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The purpose of the CTS Table 3.1.1 Required Condition A is to place the unit in a condition where RPS Instrumentation is not required to be OPERABLE. This changes the CTS by specifying the unit must be in MODE 3 instead of all OPERABLE control rods must be fully inserted. In the ITS, MODE 3 is defined as when the reactor mode switch is in the shutdown position and average reactor coolant temperature is > 212°F. When the reactor mode switch is in the shutdown position, all OPERABLE control rods will be fully inserted. Therefore since the end result is equivalent, this change is acceptable. This change is administrative because it does not result in technical changes to the CTS.

A.9 CTS Table 4.1.1 for the Mode Switch in Shutdown Instrument Channel specifies an "Operating Cycle" Frequency for the CHANNEL FUNCTIONAL TEST. CTS Table 4.1.2 for the Low Reactor Water Level transmitters, Main Steamline Isolation Valve Closure Channels, and Turbine Stop Valve Closure Instrument Channels is specifies an "Operating Cycle" Frequency for the CHANNEL CALIBRATION. CTS 1.0.F, the definition of Instrument Calibration, states that Response time is not part of the routine instrument calibration but will be checked once "per cycle." ITS SR 3.3.1.1.10 requires the performance of a CHANNEL FUNCTIONAL TEST and SR 3.3.1.1.11 requires performance of a CHANNEL CALIBRATION every "24 months." ITS SR 3.3.1.1.14 requires verification that the RPS RESPONSE TIME is within limits every "24 months" on a STAGGERED TEST BASIS. This changes the CTS by changing the Frequency from once each "Operating Cycle" to "24 months." The change to add the STAGGERED TEST BASIS allowance to ITS SR 3.3.1.1.14 is discussed in DOC L.11.

This change is acceptable because the current "Operating Cycle" is "24 months." In letter L-MT-04-036, from Thomas J. Palmisano (NMC) to the USNRC, dated June 30, 2004, NMC has proposed to extend the fuel cycle from 18 months to 24 months and at the same time has performed an evaluation in accordance with Generic Letter 91-04 to extend the unit Surveillance Requirements from 18 months to 24 months. CTS Tables 4.1.1 and 4.1.2 and the response time verification required by CTS 1.0.F were included in this evaluation. This change is designated as administrative because it does not result in any technical changes to the CTS.

A.10 CTS Table 4.1.1 Note 4 states that functional tests are not required when the systems are not required to be OPERABLE or are tripped. In addition, the Note states that if tests are missed, they shall be performed prior to returning the systems to an OPERABLE status. CTS Table 4.1.2 Note 2 includes a similar Note for calibration tests. These explicit requirements are not retained in ITS 3.3.1.1. This changes the CTS by not including these explicit requirements.

The purpose of this Note is to provide guidance on when Surveillances are required to be met and performed. This explicit Note is not needed in ITS 3.3.1.1 since these allowances are included in ITS SR 3.0.1. ITS SR 3.0.1 states that SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR, and failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. SR 3.0.1 also states that SRs are not required to be performed on inoperable equipment. When equipment is declared inoperable, the Actions of this LCO require the equipment to be placed in the trip

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condition. In this condition, the equipment is still inoperable but has accomplished the required safety function. Therefore the allowances in SR 3.0.1 and the associated actions provide adequate guidance with respect to when the associated surveillances are required to be performed and this explicit requirement is not retained. This change is administrative because it does not result in a technical change to the CTS.

A.11 CTS Table 4.1.1 Note 5 states that a functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response, alarm, and/or initiating action. These explicit requirements are not retained in ITS 3.3.1.1. This changes the CTS by not including these explicit requirements.

The purpose of CTS Table 3.1.1 Note 5 is to provide guidance on how to perform an instrument functional test of the Low Reactor Water Level, APRM/Flow Reference, and IRM instrument channels. This explicit Note is not needed in ITS 3.3.1.1 since the requirements for the CHANNEL FUNCTIONAL TEST are included in ITS 1.0, "Definitions." ITS 1.0 states that a CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. Therefore, the ITS 1.0 definition provides adequate guidance with respect to performance requirements of a CHANNEL FUNCTIONAL TEST and this explicit requirement is not retained. This change is administrative because it does not result in a technical change to the CTS.

A.12 CTS Table 3.1.1 requires the Turbine Control Valve Fast Closure and Turbine Stop Valve Closure Trip Functions (Trip Functions 11 and 12) to be OPERABLE when the reactor mode switch is in the Run position. However, CTS Table 3.1.1 footnote **.d states that these scram functions may be bypassed when the reactor thermal power is ≤ 45%. ITS Table 3.3.1.1-1 Functions 8 and 9 require the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure - Low Functions to be OPERABLE at > 45% RTP. This changes the CTS by clearly stating the Applicability of the Turbine Control Valve Fast Closure and Turbine Stop Valve Closure Trip Functions.

The purpose of the two Trip Functions Applicability is to state when they are required to be OPERABLE. Since the unit is precluded from operating at > 45% RTP without the reactor mode switch being in the Run position, there is no reason to include it in the ITS Applicability. Stating that the ITS Functions are Applicable at > 45% RTP is sufficient. Therefore, this change is acceptable. This change is administrative because it does not result in a technical change to the CTS.

A.13 When the requirements of CTS 3.1.B are not met for the Mode Switch in Shutdown, Manual Scram, Neutron Flux IRM High - High, Neutron Flux IRM Inoperable, and Scram Discharge Volume High Level (East and West) Trip Functions (CTS Table 3.1.1 Trip Functions 1, 2, 3.a, 3.b, 8.a, and 8.b), CTS Table 3.1.1 (Required Condition A) requires all OPERABLE control rods to be fully inserted. Under similar conditions in the ITS (i.e., the Required Actions and

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associated Completion Times of ACTIONS A, B, and C are not met) and when the unit is in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, ITS 3.3.1.1 ACTION H requires immediate initiation of action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. This changes the CTS by specifying the unit must initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies instead of all OPERABLE control rods must be fully inserted. The change to allow some control rods to not be inserted (those in core cells containing no fuel) is discussed in DOC L.3.

The purpose of the CTS Table 3.1.1 Required Condition A is to place the unit in a condition where RPS Instrumentation is not required to be OPERABLE. This changes the CTS by specifying the unit must initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies instead of having all OPERABLE control rods must be fully inserted. In the ITS, MODE 5 is defined when the reactor mode switch is in the Refuel position and one or more reactor vessel head closure bolts are less than fully tensioned. The Applicability has been changed to only require these RPS trip functions to be OPERABLE when a control rod is withdrawn from a core cell containing one or more fuel assemblies when the reactor mode switch is in the refuel position and one or more vessel head closure bolts are less than fully tensioned. This change is discussed in DOC L.3. Therefore, this change is acceptable. This change is administrative because it simply aligns the Applicability with the actions.

A.14 CTS Table 3.1.1 Note (1) states that there shall be two operable "or tripped" trip systems for each function. The allowance to trip a channel or trip system is included in the ITS 3.3.1.1 ACTIONS. This changes the CTS by deleting the statement requiring two "tripped" trip systems for each function. The detail that there are two trip systems has been relocated to the Bases in accordance with DOC LA.1.

This change is acceptable because ITS LCO 3.3.1.1 and Table 3.3.1.1-1 specifies the RPS Instrumentation Functions that must be OPERABLE and ITS 3.3.1.1 ACTIONS A, B, and C provide requirements for when a channel or trip system shall be placed in the trip condition. These requirements are consistent with the intent of the requirements in CTS 3.1.B and CTS Table 3.1.1, including the Notes. This change is administrative because it does not result in technical changes to the CTS.

A.15 When the requirements of CTS 3.1.B are not met for the Flow Referenced Neutron Flux APRM High-High, Inoperative, and High Flow Clamp Trip Functions (CTS Table 3.1.1 Trip Functions 4.a, 4.b, and 4.c), CTS Table 3.1.1 Required Condition A or B must be taken. When the requirements of CTS 3.1.B are not met for the Main Steamline Isolation Valve Closure Trip Function (CTS Table 3.1.1 Trip Function 10), CTS Table 3.1.1 Required Condition A or C must be taken. Required Condition A requires all OPERABLE control rods to be fully inserted, Required Condition B requires reactor power to be on the IRM range or below and the reactor to be in Startup, Refuel, or Shutdown mode, and Required Condition C requires the unit to be in Startup or Refuel mode and pressure below 600 psig. Under similar conditions in the ITS (i.e., the Required Actions and associated Completion Times of ACTIONS A, B, and C not met), ITS 3.3.1.1

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ACTION F requires the unit to be in MODE 2, and for the Main Steam Isolation Valve - Closure Function only, requires reactor pressure to be reduced to < 600 psig within 12 hours. This changes the CTS by only specifying the highest MODE that will result in the unit exiting the Applicability.

The purpose of specifying Required Conditions A, B, and C is to provide all options for exiting the Applicability of the Flow Referenced Neutron Flux APRM High-High, Inoperative, and High Flow Clamp and Main Steamline Isolation Valve Closure Trip Functions. However, it is not necessary to specify all the options; only the highest MODE allowed is required to be listed. Placing the unit in MODE 2 as required by ITS 3.3.1.1 ACTION F places the unit outside the Applicability of the APRM Functions. Placing the unit in MODE 2 and reducing reactor pressure to < 600 psig as required by ITS 3.3.1.1 ACTION F places the unit outside the Applicability of the Main Steam Isolation Valve - Closure Function. Once outside these Applicabilities, continuing to MODE 3, 4, or 5 is not precluded in the ITS. Therefore, this change is acceptable and is simply a presentation preference to be consistent with NUREG-1433, Rev. 3. This change is administrative because it does not result in technical changes to the CTS.

A.16 CTS 3.1.A states that the "setpoints" must be set in accordance with Table 3.1.1. CTS Table 3.1.1 has a column that specifies the "Limiting Trip Settings" for each RPS instrument Function. ITS LCO 3.3.1.1 requires the RPS instrumentation for each Function in Table 3.3.1.1-1 to be OPERABLE and ITS Table 3.3.1.1-1 has a column that specifies the "Allowable Value" for each Function. This changes the CTS by replacing the term "setpoints" in CTS 3.1.A and the column title "Limiting Trip Settings" in CTS Table 3.1.1 with the column title "Allowable Value" in ITS 3.3.1.1 and Table 3.3.1.1-1. Note that this change does not change the individual values in the CTS Table 3.1.1 Limiting Trip Settings column. Any changes to the individual values in the CTS Table 3.1.1 Limiting Trip Setting column is discussed in DOC L.12.

The purpose of the "Limiting Trip Settings" column in CTS Table 3.1.1 is to define the OPERABILITY limits for the RPS instrumentation Functions. Therefore, the use of the term "setpoint" in CTS 3.1.A and the title "Limiting Trip Settings" in the CTS Table 3.1.1 column is the same as the use of the title "Allowable Value" in ITS 3.3.1.1 and Table 3.3.1.1-1. The ITS Table 3.3.1.1-1 column "Allowable Value" defines the OPERABILITY limits for the individual Functions in the Table. This proposed change does not modify the actual values listed in the "Limiting Trip Settings" column in CTS Table 3.1.1 for any of the RPS Functions. Any changes to the actual value listed in the "Limiting Trip Settings" column of CTS Table 3.1.1 (i.e., changing the limit used for OPERABILITY) are discussed in DOC L.12. This change is designated as administrative change and is acceptable because it does not result in any technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 4.1.A and CTS Tables 4.1.1 and 4.1.2 do not specify requirements for a LOGIC SYSTEM FUNCTIONAL TEST. ITS Table 3.3.1.1-1 requires the performance of SR 3.3.1.1.12, a LOGIC SYSTEM FUNCTIONAL TEST every

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24 months, for each RPS Instrumentation Function. This changes the CTS by explicitly requiring a LOGIC SYSTEM FUNCTIONAL TEST to be performed on each RPS Function.

This change is acceptable because a LOGIC SYSTEM FUNCTIONAL TEST helps to ensure the RPS Instrumentation logic is functioning as required to support the safety analyses. As such, explicitly including requirements for a LOGIC SYSTEM FUNCTIONAL TEST in the Technical Specifications provides additional assurance that the OPERABILITY of the RPS Instrumentation Functions will be maintained. This change is more restrictive because it adds a specific requirement to perform a LOGIC SYSTEM FUNCTIONAL TEST on each RPS Instrumentation Function that is not currently required by the CTS.

M.2 CTS 3.1.B.1 states that with one required instrument channel inoperable in one or more trip functions, place the inoperable channel(s) or trip system in the tripped condition within 12 hours. ITS 3.3.1.1 ACTION C covers the condition of one or more Functions with RPS trip capability not maintained, and only allows one hour to restore RPS trip capability. This changes the CTS by requiring entry into ITS 3.3.1.1 ACTION C when any manual trip channel (Manual Scram and Reactor Mode Switch - Shutdown Position) is inoperable, instead of allowing 12 hours to trip the inoperable channel.

The purpose of CTS 3.1.B is to allow an inoperable RPS channel 12 hours to either place the channel in trip or place the associated trip system in trip. The 12 hour restoration time is allowed since it is assumed that the RPS can still generate a trip signal from the associated Function with only one channel inoperable. For the Manual Scram and Reactor Mode Switch -Shutdown Position Functions, this is not the case. Each RPS trip system only has one channel for each of these two Functions. Therefore, when one channel for either of these two Functions is inoperable, an RPS trip signal cannot be generated. Therefore, allowing 12 hours to restore an inoperable channel is not appropriate. ITS 3.3.1.1 ACTION C will allow only 1 hour to restore RPS trip capability in this condition. This change is acceptable since the 1 hour Completion Time will allow time to evaluate and repair any discovered inoperabilities and because it minimizes risk while allowing time for restoration or tripping a channel. This change is more restrictive because more stringent Required Actions are being applied in the ITS than were applied in the CTS.

M.3 CTS 3.1.B.3 requires the plant to be placed and maintained under the specified conditions using normal operating procedures if CTS 3.1.B.1 and CTS 3.1.B.2 are not met. CTS Table 3.1.1 Note * provides the Required Conditions when specified by CTS 3.1.B.3. CTS Table 3.1.1 Required Condition A states "All operable control rod fully inserted." CTS Table 3.1.1 Required Condition B states "Power on IRM range or below and reactor in Startup, Refuel, or Shutdown mode." CTS Table 3.1.1 Required Condition C states "Reactor in Startup or Refuel mode and pressure below 600 psig." CTS Table 3.1.1 Required Condition D states "Reactor Power less than 45%." However, no time is specified to complete the Required Conditions. ITS 3.3.1.1 ACTION E requires the plant to reduce THERMAL POWER to $\leq$ 45% RTP within 4 hours. ITS 3.3.1.1 ACTION F requires the plant to be in MODE 2 within 6 hours and to reduce reactor pressure < 600 psig within 12 hours. ITS 3.3.1.1 ACTION G

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requires the plant to be in MODE 3 in 12 hours. ITS 3.3.1.1 ACTION H requires immediate action to initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. This changes the CTS by providing specific times to reach the required conditions. Changes to the actual required condition are discussed in other DOCs.

The purpose of the proposed ACTIONS is to place the unit outside of the Applicability of each Specification. Currently times are not provided in the CTS. This change places explicit times to achieve the specified conditions. This change is acceptable since specifying Completion Times is consistent with NUREG-1433, Rev. 3. This change is more restrictive because plant operations are more limited by the ITS requirements than the CTS.

M.4 CTS Table 4.1.2 requires the performance of an APRM calibration, and Note 4 states that this calibration is performed by taking a heat balance and adjusting the APRM to agree with the heat balance. ITS SR 3.3.1.1.2 requires the verification that the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP. This changes the CTS by adding an explicit acceptance criterion for the test (i.e., ≤ 2% RTP).

The purpose of CTS Table 4.1.2 including Note 4 is to ensure the APRMs are calibrated, ensuring they will function properly to mitigate the consequences of a design basis accident or transient. This change adds an acceptance criterion that the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP. The proposed acceptance criterion is acceptable since the design basis accident and overpressure protection analyses are normally performed at a THERMAL POWER of 2% greater than RATED THERMAL POWER. This change is more restrictive because it adds an explicit acceptance criteria to the APRM calibration Surveillance, entering MODE 2 from MODE 1.

M.5 CTS Table 3.1.1 requires the High Reactor Pressure Trip Function (Trip Function 5) to be OPERABLE when the reactor mode switch is in the Refuel, Startup, and Run position. However, CTS Table 3.1.1 Note (9) states that the Trip Function is not required to be OPERABLE when the reactor vessel head is unbolted (i.e., one or more reactor head closure bolts less than fully tensioned). Furthermore, CTS Table 3.1.1 Note (3) states that the only RPS Trip Functions that are required to be OPERABLE when in the refueling mode with the reactor subcritical and reactor water temperature less than 212°F are Mode Switch in Shutdown, Manual Scram, High Flux IRM (i.e., Neutron Flux IRM High - High and Neutron Flux IRM Inoperative), and Scram Discharge Volume High Level. ITS Table 3.3.1.1-1 requires the Reactor Vessel Steam Dome Pressure - High Function to be OPERABLE in MODES 1 and 2. This changes the CTS by requiring the High Reactor Pressure Function to be OPERABLE at all times when the reactor mode switch is in the Startup/Hot Standby position and the Run positions regardless of the status of the reactor vessel head bolts.

The purpose of CTS Table 3.1.1 is to ensure the High Reactor Pressure Function is OPERABLE when necessary to mitigate the consequences of a transient or design basis accident. This change requires the High Reactor Pressure Function to be OPERABLE at all times when the reactor mode switch is in the Startup/Hot

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Standby position and the Run positions, regardless of the status of the reactor vessel head bolts. The reactor mode switch is not normally placed in the Startup/Hot Standby or Run position with the High Reactor Pressure Function inoperable unless the reactor mode switch interlocks functions are being tested. ITS 3.10.2 covers testing requirements for the reactor mode switch, and allows the reactor mode switch position specified in ITS Table 1.1-1 for MODES 3, 4, and 5 to be changed to include the run, startup/hot standby, and refuel position, and operation considered not to be in MODE 1 or 2, to allow testing of instrumentation associated with the reactor mode switch interlock functions, provided all control rods remain fully inserted in core cells containing one or more fuel assemblies and no CORE ALTERATIONS are in progress. These additional requirements will help ensure there is no challenge to the RPS System because all control rods will be fully inserted into a core cell containing one or more fuel assemblies. This change is acceptable because the ITS will specifically require the High Reactor Pressure Function to be OPERABLE when the reactor head is unbolted and the reactor mode switch is in the Run or Startup/Hot Standby positions, unless the special operations LCO is being followed. This change is more restrictive because the ITS Applicability of the High Reactor Pressure Function covers more conditions than the CTS Applicability.

M.6 CTS Table 3.1.1 requires the High Drywell Pressure Trip Function (Trip Function 6) to be OPERABLE when the reactor mode switch is in the Refuel, Startup, and Run positions. However, CTS Table 3.1.1 Note (4) states that this Function is not required to be OPERABLE when primary containment integrity is not required. CTS 3.7.A.2.a.(1) requires the primary containment integrity to be applicable at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel. Furthermore, CTS Table 3.1.1 Note (3) states that the only RPS Trip Functions that are required to be OPERABLE when in the refueling mode with the reactor subcritical and reactor water temperature less than 212°F are Mode Switch in Shutdown, Manual Scram, High Flux IRM (i.e., Neutron Flux IRM High - High and Neutron Flux IRM Inoperative), and Scram Discharge Volume High Level.
ITS Table 3.3.1.1-1 Function 6 requires the Drywell Pressure - High Function to be OPERABLE in MODES 1 and 2. This changes the CTS by requiring the High Drywell Pressure Trip Function to be OPERABLE at all times when the reactor mode switch is in the Refuel and Startup positions when the vessel head is on, even if the reactor is subcritical and temperature is below 212°F.

The purpose of CTS Table 3.1.1 Trip Function 6 is to ensure the High Drywell Pressure Trip Function is OPERABLE when necessary to mitigate the consequences of a design basis accident. The High Drywell Pressure Trip Function is required to be OPERABLE in MODES 1 and 2 because there may be considerable energy in the reactor coolant system and if there is an accident this RPS Function may be necessary to mitigate the consequences of a design basis event. Currently, the Function is not required when the reactor is not critical and temperature is below 212°F. In MODE 2, the reactor coolant temperature may be less than or equal to 212°F when the reactor is subcritical but control rods are withdrawn. Therefore, it is necessary and acceptable to require the High Drywell Pressure Trip Function to be OPERABLE. This change is more restrictive because the LCO will be applicable under more reactor operating conditions than in the CTS.

M.7 CTS Table 3.1.1 requires the Reactor Low Water Level Trip Function (Trip Function 7) to be OPERABLE when the reactor mode switch is in the Refuel, Startup, and Run positions. However, CTS Table 3.1.1 Note (3) states that when the reactor mode switch is in the refuel position and the reactor is subcritical and reactor water temperature is less than 212°F the only RPS Trip Functions that are required to be OPERABLE are Mode Switch in Shutdown, Manual Scram, High Flux IRM (i.e., Neutron Flux IRM High - High and Neutron Flux IRM Inoperative), and Scram Discharge Volume High Level. ITS Table 3.3.1.1-1 Function 4 requires the Reactor Vessel Water Level - Low Function to be OPERABLE in MODES 1 and 2. This changes the CTS by requiring Reactor Low Water Level Trip Function to be OPERABLE when the reactor mode switch is in the Refuel position and the vessel head is on, even if the reactor is subcritical and temperature is below 212°F.

The purpose of CTS Table 3.1.1 Trip Function 7 is to ensure the Reactor Low Water Level Trip Function is OPERABLE when necessary to mitigate the consequences of a transient or design basis accident, as applicable. The Reactor Low Water Level Trip Function is required to be OPERABLE in MODE 2 (i.e., the reactor mode switch in the refuel position and all reactor vessel head closure bolts are fully tensioned) because when the vessel head is on and the closure bolts are fully tensioned, the reactor is ready to begin power operation. This change is acceptable because it ensures the RPS instrumentation is available when the unit is prepared to begin power operation. This change is more restrictive because the LCO will be applicable under more reactor operating conditions than in the CTS.

M.8 CTS Table 3.1.1 Note (e) allows the High Drywell Pressure Trip Function (Trip Function 6) to be bypassed in Startup and Run modes during purging for containment inerting or de-inerting operations by closing the manual containment isolation valves. ITS Table 3.3.1.1-1 does not include this bypass allowance for the Drywell Pressure - High Function (Function 6). This changes the CTS by deleting the allowance to bypass the High Drywell Pressure Trip Function during containment purging operations.

The purpose of this Note is to allow the High Drywell Pressure Trip Function to be bypassed to avoid an inadvertent scram during the purging operations. The Monticello plant does not utilize this High Drywell Pressure Trip Function bypass allowance during any type of purging operation. The allowance has been deleted from the Technical Specifications. This change is acceptable because this allowance is not needed for purging operations. This change is more restrictive because the bypass allowance has been deleted from the Technical Specifications.

M.9 CTS Table 3.1.1 Trip Function 11 (Turbine Control Valve Fast Closure) references Note (7) for the Limiting Trip Setting. However, Note (7) states that the trip is upon a loss of oil pressure to the acceleration relay. No specific oil pressure is provided. ITS Table 3.3.1.1-1 Function 9 title includes the information concerning low oil pressure to the acceleration relay and specifies

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the Allowable Value for this Function to be ≥ 167.8 psig. This changes the CTS by providing a specific value for the Allowable Value for the Turbine Control Valve Fast Closure Function.

This change is acceptable because the Allowable Value is necessary to support the OPERABILITY of the Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure - Low Function. As such, including the Allowable Value in the Technical Specifications provides additional assurance that the OPERABILITY of the Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure - Low Function is maintained. The addition of the Allowable Value is acceptable since these requirements are currently administratively controlled in procedures. This change is more restrictive because it adds explicit Allowable Values for the Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure - Low Function to the CTS.

M.10 CTS Table 4.1.1 does not provide a Surveillance to perform a functional test of each RPS automatic scram contactor every 7 days. ITS SR 3.3.1.1.4 requires a functional test of each RPS automatic scram contactor and is required for each automatic RPS Function in ITS Table 3.3.1.1-1. This changes the CTS by adding this SR associated with the automatic scram contactors.

The purpose of this test is to be consistent with the analyses of NEDC-30851P, "Technical Specification Improvement Analysis for BWR Reactor Protection System," which was approved by the NRC in an SER dated July 15, 1987. The Surveillance Test Frequencies associated with the functional test for the automatic scram Functions of CTS Table 4.1.1 were extended in Amendment 81 based on this analysis. This analysis requires testing of the automatic scram contactors every 7 days. This change is acceptable since it is consistent with NEDC-30851P and current plant practice. This change is more restrictive because it adds an explicit requirement to test the automatic scram contactors every 7 days.

M.11 CTS Table 4.1.1 does not provide a Surveillance to perform a verification that the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure - Low channels are not bypassed when THERMAL POWER is > 45% RTP every 24 months. ITS SR 3.3.1.1.13 includes this testing requirement. This changes the CTS by adding this SR associated Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure - Low channels.

The purpose of ITS SR 3.3.1.1.13 is to ensure the automatic bypass of the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure - Low channels are OPERABLE. This test has been added to the Technical Specifications. This change is acceptable since it ensures the RPS scram associated with the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure - Low channels are OPERABLE when power is > 45% RTP. This change is more restrictive because it adds an explicit requirement to test the RPS Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure - Low channels are not bypassed when power is > 45% RTP.

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M.12 CTS Table 4.1.1 Note 2 requires the performance of a sensor check of the low reactor water level channels once per day. ITS SR 3.3.1.1.1 requires the performance of a CHANNEL CHECK every 12 hours. This changes the CTS by changing the Frequency of testing from once per "day" to "12 hours."

The purpose of CTS Table 4.1.1 Note 2 is to help ensure the RPS level instrumentation channels are OPERABLE. The change is acceptable since it will continue to help ensure the channels are OPERABLE. This change is consistent with NUREG-1433, Rev.3. This change is more restrictive because the ITS will require the Surveillance to be performed more frequently than in the CTS.

M.13 CTS Table 4.1.1 Note 3 applies to the IRM channels and it requires the performance of a functional test "prior to every startup." ITS SR 3.3.1.1.3 requires the performance of a CHANNEL FUNCTIONAL TEST every 7 days. A Note is included which states that the Surveillance is not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. This changes the CTS by modifying the Frequency for the CHANNEL FUNCTIONAL TEST of the IRM channel from "prior to every startup" to "every 7 days" which will essentially require the CHANNEL FUNCTIONAL TEST to be performed during both startups and shutdowns, and adds the Note to allow entry into MODE 2 from MODE 1 to properly perform the test during a shutdown.

The purpose of CTS Table 4.1.1 is to ensure the IRM channels are OPERABLE. Currently the functional test is only required to be performed "prior to every startup." The new requirement will require the Surveillance to be performed within 7 days of entering MODE 2 or MODE 5 with any control rod withdrawn from a core cell containing one of more fuel assemblies, however the Surveillance will allow the plant to enter MODE 2 from MODE 1 and allow 12 hours to complete the Surveillance. This is necessary since the IRMs are not required to be OPERABLE in MODE 1 and performing the test in MODE 1 will require the utilization of jumpers, lifted leads, or removable links. This change is acceptable since the proposed Surveillance and Surveillance Frequency are consistent with the reliability analysis in NEDC-30851P, "Technical Specification Improvement Analysis for BWR Reactor Protection System," and approved by the NRC in an SER dated July 15, 1987. This change is designated as more restrictive because it adds a requirement to test the IRM channels "every 7 days" instead of just "prior to startup" and also requires the test to be performed within 12 hours of entering MODE 2 from MODE 1.

M.14 CTS Table 4.1.2 provides a requirement to perform a calibration of the APRMs by performing a heat balance. ITS 3.3.1.1 adds two additional Surveillances for the APRMs channels. ITS SR 3.3.1.1.6 requires the calibration of the local power range monitors every 2000 effective full power hours and ITS SR 3.3.1.1.9 requires the performance of a CHANNEL CALIBRATION of the APRM channel every 92 days. However, ITS SR 3.3.1.1.9 is modified by a Note that states "Neutron detectors are excluded." This changes the CTS by adding two new Surveillances to ensure the APRM channels are operating properly.

The purpose of ITS SR 3.3.1.1.6 and SR 3.3.1.1.9 is to ensure the APRM channels remain OPERABLE. LPRMs provide input into the APRM channels therefore a calibration is necessary to ensure this portion of the channel remains

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OPERABLE. CTS Table 4.1.2 requires the output of the APRM channels to be adjusted to be consistent with the heat balance and it also requires a calibration of the flow channels. The heat balance Surveillance test does not ensure the entire APRM flux portion of the channel is calibrated and the flow channel calibration does not ensure the flow channels are calibrated to conform to core flow readings. The proposed testing is necessary to help ensure the entire channel is OPERABLE. The allowance in ITS SR 3.3.1.1.9 to exclude neutron detectors is necessary because neutron detectors are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal and since changes in neutron detector sensitivity are compensated for by performing the 7 day heat balance calibration (ITS SR 3.3.1.1.2). This change is acceptable because it helps to ensure the APRM channel remain OPERABLE. This change is more restrictive because it adds two new Surveillance Requirements.

M.15 CTS 4.1.1 does not provide any requirements to perform a CHANNEL CHECK on the IRM and APRM/Flow Reference instrument channels. ITS Table 3.3.1.1-1 Function 1.a (Intermediate Range Monitors Neutron Flux - High High) and Function 2.a (Average Power Range Monitors Flow Referenced Neutron Flux - High High) require the performance of SR 3.3.1.1.1, a CHANNEL CHECK, every 12 hours. This changes the CTS by explicitly requiring a CHANNEL CHECK to be performed on the IRM and APRM channels.

This change is acceptable because a CHANNEL CHECK helps to ensure these RPS Functions are indicated correctly which helps to ensure they are OPERABLE to support the safety analyses. As such, explicitly including requirements for a CHANNEL CHECK in the Technical Specifications provides additional assurance that the OPERABILITY of these RPS Trip Functions will be maintained. This change is more restrictive because it adds explicit requirements to perform a CHANNEL CHECK for IRM (ITS Table 3.3.1.1-1 Function 1.a) and APRM (ITS Table 3.3.1.1-1 Function 2.a) RPS Trip Functions.

RELOCATED SPECIFICATIONS

R.1 CTS 3.1.A requires the RPS Turbine Condenser Low Vacuum Trip Function (CTS Table 3.1.1 Trip Function 9) to be OPERABLE while CTS 4.1.A requires the RPS Turbine Condenser Low Vacuum Trip Function channels to be functional tested and calibrated as indicated in Table 4.1.1 and 4.1.2, respectively. The turbine condenser low vacuum scram is provided to protect the main condenser from overpressurization in the event that vacuum is lost. A loss of condenser vacuum would cause the turbine stop valves to close, resulting in a turbine trip transient. The low condenser vacuum trip anticipates this transient and scrams the reactor. No design basis accidents or transients take credit for this scram signal. This Specification does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual (TRM).

This change is acceptable because the requirements of CTS 3.1.A and CTS 4.1.A related to the Turbine Condenser Low Vacuum trip function do not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

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10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The Turbine Condenser Low Vacuum scram instrumentation is not an instrument used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA). The Turbine Condenser Low Vacuum trip function does not satisfy criterion 1.

2. The Turbine Condenser Low Vacuum scram instrumentation is not a process variable 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. The Turbine Condenser Low Vacuum trip function does not satisfy criterion 2.

3. The Turbine Condenser Low Vacuum scram instrumentation 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. The Turbine Condenser Low Vacuum trip function does not satisfy criterion 3.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (item 337) of NEDO-31466, Supplement 1, the loss of the Turbine Condenser Low Vacuum scram instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. Nuclear Management Company, LLC has reviewed this evaluation, considers it applicable to Monticello, and concurs with the assessment.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the Turbine Condenser Low Vacuum LCO, Actions, and associated Surveillances may be relocated out of the Technical Specifications. The Turbine Condenser Low Vacuum Specification will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the LCO did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

LA.1 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.1.A states that the RPS response time "shall not exceed 50 milliseconds." ITS SR 3.3.1.1.15 requires the RPS RESPONSE TIME to be "within limits." This changes the CTS by relocating the details of the actual response time limit to the ITS Bases.

The removal of this detail, which is related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that the RPS RESPONSE time must remain within limit. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases.

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Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is consistent with NRC Generic Letter 93-08 dated December 30, 1993 that provided guidance on relocating the Technical Specification Table of Instrument Response Time Limits. This change is a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA.2 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.1.A states that the "minimum number of trip systems" that must be OPERABLE for the RPS Instrumentation is in Table 3.1.1 and CTS 3.1.B provides a Condition for an inoperable trip system, however no explicit actions are provided for any inoperable trip system. CTS Table 3.1.1 Note (1) states that there shall be two operable or tripped trip systems for each function. In addition, CTS Table 3.1.1 provides a requirement for the "Total No. of Instrument Channels Per Trip System" for each RPS Instrumentation Trip Function. ITS 3.3.1.1 does not include these details. This changes the CTS by moving the information of the required number of OPERABLE trip systems and the "Total No. of Instrument Channels per Trip System" to the ITS Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the minimum number of required channels for each trip system. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA.3 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.1.1 states that the Neutron Flux IRM - High High (Trip Function 3.a) Limiting Trip Setting is $\leq$ 120/125 of full scale AND "< 20% of Rated Thermal Power." The CTS Table 3.1.1 Flow Referenced Neutron Flux APRM High High (Trip Function 4.a) Limiting Trip Setting provides a flow reference setting as a function of "W" and states "W=percent of recirculation drive flow to produce a core flow of 57.6×10^6 lbm/hr." ITS Table 3.3.1.1-1 Function 1.a provides the Allowable Value for the IRM Neutron Flux - High High Function, but does not include the "< 20% of Rated Thermal Power" portion of the Allowable Value. ITS Table 3.3.1.1-1 Function 2 provides the flow referenced equation for the APRM Flow Reference Neutron Flux - High High Function, however the definition of "W" is not retained. This changes the CTS by moving the details of the definition of "W" and the "20% of Rated Thermal Power" requirement to the ITS Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the Allowable Values associated with these Functions. In addition, the "< 20% of Rated Thermal Power" portion of the IRM Neutron Flux - High High Function Allowable Value is a basis for the 120/125 of full scale Allowable Value. The 120/125 of full scale Allowable Value ensures that the IRM Neutron Flux - High High Functions will trip prior to reactor power exceeding 20% RTP; it is not an actual Allowable Value, since the IRM does not directly monitor Thermal Power nor does it indicate Thermal Power. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA.4 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS Table 3.1.1 Note (2) states that a Neutron Flux IRM (Trip Function 3) channel is considered to be OPERABLE if "its detector is fully inserted." CTS Table 3.1.1 Note (5) states that a Flow Referenced Neutron Flux APRM (Trip Function 4) channel is considered to be operable if "2 LPRM inputs per level and at least a total of 14 LPRM inputs, except that channels 1, 2, 5, and 6 may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable." ITS 3.3.1.1 does not include these details. This changes the CTS by relocating the details for meeting TS requirements to the ITS Bases.

The removal of these details for meeting TS requirements from the CTS is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that the Average Power Range Monitors and the Intermediate Range Monitors must be OPERABLE. Also, this change is acceptable because these types of details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the CTS.

LA.5 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS Table 3.1.1 Trip Function 8 provides the Limiting Trip Setting for the Scram Discharge Volume High level channels. CTS Table 3.1.1 Note (8) states that the Limiting Trip Setting for the Scram Discharge Volume High Level channels refers to the volume of water in the discharge volume receiver tank and does not include the volume in the lines to the level switches. ITS Table 3.3.1.1-1 Function 7 provides the Allowable Value for this Function and does not include these details. This changes the CTS by relocating the details for meeting TS requirements to the ITS Bases.

The removal of these details for meeting TS requirements from the CTS is acceptable because this type of information is not necessary to be included in the

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Technical Specifications to provide adequate protection of public health and safety. The ITS still retains an Allowable Value for the Scram Discharge Volume Water Level - High Function. Also, this change is acceptable because these types of details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the CTS.

LA.6 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 4.1.2 defines the "Group" each RPS Instrumentation Function is assigned. The Table defines two groups. Group A is defined as "Passive type devices" and Group B is defined as "Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity." ITS 3.3.1.1 does not include this information. This changes the CTS by relocating the design details associated with the RPS Instrumentation "Groups" to the USAR.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirements for the Functions to be OPERABLE and tested in accordance with the assigned Surveillance Requirements. Also, this change is acceptable because the removed information will be adequately controlled in the USAR. The USAR is controlled under 10 CFR 50.59 or 10 CFR 50.71(e), which ensures changes are properly evaluated. This change is a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA.7 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS Table 4.1.2 specifies that the calibration method for the APRM test is a heat balance. CTS Table 4.1.2 Note 4 also states that this test is performed by taking a heat balance and adjusting the APRM to agree with the heat balance. ITS SR 3.3.1.1.2 does not include the method for performing the Surveillance. This changes the CTS by relocating the procedural details for meeting TS requirements to the ITS Bases.

The removal of these details for meeting TS requirements from the CTS is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement that the Average Power Range Monitor must be adjusted to conform to the calculated power. Also, this change is acceptable because these types of procedural details will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the CTS.

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LESS RESTRICTIVE CHANGES

L.1 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.1.A states that the initiation of "any" channel trip to the de-energization of the scram pilot valve solenoids shall not exceed 50 milliseconds. This is essentially a response time requirement. ITS SR 3.3.1.1.14 requires the verification of the RPS RESPONSE TIME. ITS Table 3.3.1.1-1 requires the RPS RESPONSE TIME test to be performed on certain RPS Functions, but not all RPS Functions. This changes the CTS by requiring the testing to be performed only on certain Functions.

The purpose of the CTS 3.1.A is to ensure the RPS channels respond within appropriate time to help ensure the safety analyses assumptions are met. The IRM Inop, APRM Inop, Drywell Pressure - High, Scram Discharge Volume Water Level - High, and the manual scram Functions (Reactor Mode Switch - Shutdown Position and Manual Scram) are not credited in the safety analyses, and therefore the proposed RPS RESPONSE TIME test (ITS SR 3.3.1.1.14) is only associated with those Functions that are credited in the accident analysis where an explicit RPS RESPONSE TIME is assumed. This change is acceptable since the OPERABILITY of the channels will still be confirmed during the LOGIC SYSTEM FUNCTIONAL TEST, CHANNEL FUNCTIONAL TEST, and the CHANNEL CALIBRATION Surveillances, as applicable. This change is less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS.

L.2 *(Category 4 - Relaxation of Required Action)* When more than one instrument channel is inoperable for one or more trip functions, CTS 3.1.B.2 requires the immediate placement of the appropriate channel(s) or trip system(s) in the tripped condition. ITS 3.3.1.1 ACTION A covers the situation when one or more required channels are inoperable, and allows 12 hours to either place the channel in trip or to place the associated trip system in trip. ITS 3.3.1.1 ACTION B covers the condition for one or more Functions with one or more required channels inoperable in both trip systems, and requires either the placement of the inoperable channel in one trip system in trip or the placement of one trip system in trip within 6 hours. ITS 3.3.1.1 ACTION C covers the condition for one or more Functions with RPS trip capability not maintained, and allows one hour to restore RPS trip capability. This changes the CTS by allowing 6 hours to take action (by either restoring or tripping a channel) when one or more Functions have one or more required channels inoperable in both trip systems and allowing 1 hour to restore automatic RPS trip capability (by either restoring or tripping a channel) when one or more Functions have two channels in a trip system inoperable (i.e., it is not maintaining RPS trip capability) instead of requiring immediate action to be taken.

The purpose of the CTS 3.1.B.2 is to ensure that each inoperable RPS Function channel or associated trip system is immediately placed in trip when more than one instrument channel in a Function is found to be inoperable. ITS 3.3.1.1 ACTION B will allow 6 hours to place a channel in trip under the same circumstances. However, an additional restriction has been included that requires the restoration of RPS trip capability whenever it is lost within one hour. This change is acceptable since within the 6 hour time, the associated inoperable Function will have the Function in at least a condition equivalent to CTS 3.1.B.1.

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Completing one of the Required Actions restores RPS to a reliability level equivalent to that evaluated in the reliability analysis of NEDO-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988, which was used to justify the 12 hour allowance in CTS 3.1.B.1 as previously approved by the NRC in Amendment 81. The 6 hour Completion time is acceptable based on the remaining capability to trip, the diversity of the sensors available to provide the trip signal, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram. The 1 hour Completion Time in ITS 3.3.1.1 ACTION C is acceptable since it will allow time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping channels. This change is less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.3 *(Category 2 – Relaxation of Applicability)* CTS Table 3.1.1 requires the Mode Switch in Shutdown, Manual Scram, Neutron Flux IRM High - High, Neutron Flux IRM Inoperative, Scram Discharge Volume High Level (East and West) Trip Functions (CTS Table 3.1.1 Trip Functions 1, 2, 3.a, 3.b, 8.a, and 8.b, respectively) to be OPERABLE when the reactor mode switch is in the Refuel, Startup, and Run (for Trip Functions 1, 2, 8.a, and 8.b only) positions. Furthermore, CTS Table 3.1.1 Note (3) states that these Functions are the only RPS Trip Functions that are required to be OPERABLE when in the refueling mode with the reactor subcritical and reactor water temperature less than 212°F. (The Note 3 reference to High Flux IRM refers to both the Neutron Flux IRM High - High and Inoperative Functions.) CTS Table 3.1.1 footnote **.a allows the Scram Discharge Volume High Level Trip Function to be bypassed in the Refuel mode. During this time, a control rod block is inserted. ITS Table 3.3.1.1-1 requires these Functions to be OPERABLE during MODES 1 (Functions 7.a, 7.b, 10, and 11 only) and 2 and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies (Table 3.3.1.1-1 Footnote (a)). This changes the CTS by only requiring these RPS Trip Functions to be OPERABLE when the reactor mode switch is in the refuel position and one or more vessel head closure bolts are less than fully tensioned (i.e., MODES) only when a control rod is withdrawn from a core cell containing one or more fuel assemblies.

The purpose of CTS Table 3.1.1 is to ensure the appropriate RPS Trip Functions are OPERABLE when necessary to mitigate the consequences of a transient or design basis accident. This change is acceptable because the requirements continue to ensure that the structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. Currently the specified Functions are required to be OPERABLE when the reactor mode switch is in the Refuel, Startup, and Run (for Trip Functions 1, 2, 8.a, and 8.b only) positions. Furthermore, CTS Table 3.1.1 Note (3) states that these Functions are the only RPS Trip Functions that are required to be OPERABLE when in the refueling mode with the reactor subcritical and reactor water temperature less than 212°F. In the ITS, MODE 1 is when the reactor mode switch is in the Run position while MODE 2 covers the situations when the reactor mode switch is in the Startup/Hot Standby position and the Refuel position when all reactor vessel head closure bolts are fully tensioned. In

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addition, MODE 5 covers the situation when the reactor mode switch is in the Refuel position and one or more reactor vessel head closure bolts are less than fully tensioned. The proposed Applicability covers all the conditions specified in the CTS except that in MODE 5 the RPS Trip Functions are only required to be OPERABLE with any control rod withdrawn from a core cell containing one or more fuel assemblies. This change is acceptable because control rods withdrawn from or inserted into a core cell containing no fuel assemblies have a negligible impact on the reactivity of the core and therefore are not required to be OPERABLE with the capability to scram. Provided all rods otherwise remain inserted, the RPS Functions serve no purpose and are not required. In this condition the required SHUTDOWN MARGIN (ITS 3.1.1) and the required one-rod-out interlock (ITS 3.9.2) ensure no event requiring the RPS will occur. In addition, due to this change, the allowance to bypass the Scram Discharge Volume High Level Trip Function is not needed, and the rod block requirements are discussed in ITS 3.3.2.1. This change is less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

L.4 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS Table 3.1.1 footnote **.f provides guidance for the bypass of certain RPS instrument channels, and states "One instrument channel for the functions indicated in the table to allow completion of surveillance testing, provided that: 1) Redundant instrument channels in the same trip system are capable of initiating the automatic function and are demonstrated to be operable either immediately prior or immediately subsequent to applying the bypass; and 2) while the bypass is applied, surveillance testing shall proceed on a continuous basis and the remaining instrument channels initiating the same function are tested prior to any other. Upon completion of surveillance testing, the bypass is removed." ITS Table 3.3.1.1 does not include this Note. This changes the CTS by deleting CTS Table 3.1.1 footnote **.f.

The purpose of CTS Table 3.1.1 footnote **.f is to provide requirements on the bypass of RPS instrument channels during Surveillance testing. This change deletes these requirements from the Technical Specifications. CTS Table 3.1.1 Note (1) states "A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided that at least one other operable channel in the same trip system is monitoring that parameter." The guidance in CTS Table 3.1.1 Note (1) is consistent with the first part of CTS Table 3.1.1 footnote **.f except that CTS Table 3.1.1 Note (1) does not provide explicit guidance for the testing of other channels. The requirements of CTS Table 3.1.1 Note (1) are retained in ITS 3.3.1.1 Surveillance Requirements Note 2 as modified by DOC A.6. CTS Table 3.1.1 Note (1) states that a channel may be placed in an inoperable status. This inoperable status means that the channel may be bypassed. This requirement in CTS Table 3.1.1 footnote **.f to test the other automatic instrument channels "immediately prior or immediately subsequent to applying the bypass" and the requirements in CTS Table 3.1.1 footnote **.f.2 to test "the remaining instrument channels initiating the same function are tested prior to any other" have been deleted. They are overly restrictive and are not necessary. The most common outcome of the performance of Surveillances is the successful demonstration that the acceptance criteria are satisfied and OPERABILITY is verified. As long as the Surveillance Frequencies of the other

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channels are current and they are not known to be inoperable, they should be considered to be OPERABLE and such explicit requirements in TS is not necessary. CTS Table 3.1.1 Note (1) was added to the TS in Licensing Amendment 81 based on the allowances in NEDC-30851P, Technical Specification Improvement Analysis for BWR Reactor Protection System," and approved by the NRC in an SER dated July 15, 1987. This change is acceptable because the allowance in proposed ITS 3.3.1.1 Surveillance Requirements Note 2 provides sufficient restrictions for performing Surveillance Tests. This change is less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.5 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS Table 4.1.1 specifies the requirements for the functional test of various RPS Functions. The functional test requires testing of the "trip channel" and "alarm" for the High Reactor Pressure, High Drywell Pressure, Low Reactor Water Level, High Water Level in Scram Discharge Volume, Main Steam Line Isolation Valve Closure, Turbine Stop Valve Closure, Manual Scram, Turbine Control Valve Fast Closure, and IRM channels, and the functional test requires testing of the "trip output relays" for the APRM/Flow Reference channels and requires the actual placement of the mode switch in the shutdown position for the Mode Switch in Shutdown channels. CTS Table 4.1.2 Note 4 states APRM channel alarms and trips will be verified and calibrated if necessary during functional testing. ITS SR 3.3.1.1.3, SR 3.3.1.1.4, SR 3.3.1.1.7, and SR 3.3.1.1.10 require the performance of a CHANNEL FUNCTIONAL TEST, but do not specify any specific requirements for the test. This changes the CTS by deleting the specific channel functional test requirements to test the "Trip Channel and Alarm," "Trip Output relays," and "Place mode switch in shutdown."

The purpose of CTS Table 4.1.1 is to provide the appropriate channel functional test requirements for the RPS Functions. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. The definition of CHANNEL FUNCTIONAL TEST provides the appropriate guidance for testing. A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps. The requirement to perform a CHANNEL FUNCTIONAL TEST and the definition of CHANNEL FUNCTIONAL TEST provide sufficient guidance for performing the test. This change is less restrictive because the explicit requirement to test the specific component (e.g., alarms) during the CHANNEL FUNCTIONAL TEST have been deleted from the TS.

L.6 *(Category 5 – Deletion of Surveillance Requirement)* CTS Table 4.1.1 Note 3 applies to the IRM channels and requires a demonstration that the IRM and APRM channels overlap at least 1/2 decade prior to every normal shutdown. This test is not included in ITS 3.3.1.1. This changes the CTS by deleting the IRM/APRM overlap test.

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The purpose of CTS Table 4.1.1 Note 3 is to ensure there is appropriate overlap between the APRM and IRM instrumentation to ensure all power levels of the core are properly monitored by the nuclear instrumentation prior to leaving the APRM monitoring range. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO can perform its required functions. The requirements to perform the IRM/APRM overlap test is not included in the ITS. The change is acceptable since the proposed Surveillances for the IRMs (ITS SR 3.3.1.1.1, SR 3.3.1.1.3, SR 3.3.1.1.5, SR 3.3.1.1.11, and SR 3.3.1.1.12) and APRMs (ITS SR 3.3.1.1.1, SR 3.3.1.1.2, SR 3.3.1.1.5, SR 3.3.1.1.6, SR 3.3.1.1.7, SR 3.3.1.1.9, SR 3.3.1.1.12, and SR 3.3.1.1.14) will ensure that the equipment will perform their safety function. The ITS will require a CHANNEL CHECK, a CHANNEL FUNCTIONAL TEST, a CHANNEL CALIBRATION, and a LOGIC SYSTEM FUNCTIONAL TEST to be performed on each IRM channel. These Surveillances will help ensure the IRMs are capable of monitoring the core and trip under the appropriate conditions. This change is less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L.7 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS Table 4.1.2 requires the performance of an APRM calibration once every 3 days. CTS Table 4.1.2 Note 4 states that this calibration is performed by taking a heat balance and adjusting the APRM to agree with the heat balance. ITS SR 3.3.1.1.2 requires the verification that the absolute difference between the average power range monitor (APRM) channels and the calculated power is $\leq$ 2% RTP every 7 days. This changes the CTS by extending the Frequency of testing from 3 days to 7 days. The change adding in an acceptance criterion is discussed in DOC M.4.

The purpose of the CTS Table 4.1.2 APRM calibration test is to ensure the APRMs are accurately indicating the true core power, which is affected by the LPRM sensitivity. This change is acceptable because the new Surveillance Frequency will provide an acceptable level of equipment reliability. This change extends the frequency of testing from every 3 days to every 7 days. The 7 day Surveillance Frequency is acceptable, based on operating experience and the fact that only minor changes in LPRM sensitivity occur during this time frame. This change is consistent with NUREG-1433, Revision 3. This change is less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.8 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS Table 4.1.2 requires the performance of an APRM calibration once every 3 days. CTS Table 4.1.2 Note 4 states that this calibration is performed by taking a heat balance and adjusting the APRM to agree with the heat balance. ITS SR 3.3.1.1.2 requires the verification that the absolute difference between the average power range monitor (APRM) channels and the calculated power is $\leq$ 2% RTP every 7 days. A Note to ITS SR 3.3.1.1.2 states that the Surveillance is not required to be performed until 12 hours after THERMAL POWER $\geq$ 25% RTP. This changes the CTS by allowing the plant to enter MODE 1 without meeting the 7 day Frequency and adding the explicit time restraint to complete the test within 12 hours of exceeding 25% RTP. The

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change to the Frequency is discussed in DOC L.7 and the change adding in an acceptance criteria is discussed in DOC M.4.

The purpose of the ITS SR 3.3.1.1.2 Note is to allow the plant to enter the Applicability of the Specification and allow the Surveillance to be performed 12 hours after THERMAL POWER ≥ 25% RTP. This change is acceptable because the new Surveillance Frequency provides an acceptable level of equipment reliability. This exception is necessary to allow a normal startup to be completed and at the same time to allow time to perform the Surveillance. The proposed Surveillance Note provides a finite time in which the Surveillances must be performed after entering the specified condition and therefore this change is considered acceptable. A restriction is provided that requires the SR to be performed only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). This change is less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.9 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS Table 4.1.2 provides the calibration method (i.e., heat balance, pressure standard, water level, or observation) for each RPS Instrument Channel. ITS 3.3.1.1 does not include this information in the associated SRs. This changes the CTS by deleting the calibration method from the CTS.

The purpose of the calibration method is to define how the channel must be calibrated. This change deletes the calibration method from the CTS. The change is acceptable since the definition of CHANNEL CALIBRATION provides the appropriate guidance for performing a calibration. ITS 1.1 definition of CHANNEL CALIBRATION states "A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps." The definition is not explicit on how the channel must be calibrated, but there is guidance that states a CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. This definition will help ensure the channels are calibrated correctly. The procedures will include the type of calibration method since this will be necessary to satisfy the plant setpoint methodology. This change is less restrictive because it deletes the calibration method from the CTS.

L.10 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS Table 4.1.2 Note 1 requires the performance of an IRM calibration during every startup and normal shutdown. ITS Table 3.3.1.1 Function 1 (IRM)

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requires the performance of SR 3.3.1.1.11, a CHANNEL CALIBRATION, every 24 months. In addition, Note 2 allows the test to be delayed until 12 hours after entering MODE 2 from MODE 1. This changes the CTS by changing the Frequency for an IRM calibration from every startup and normal shutdown to 24 months and allows the Surveillance to be delayed during a shutdown until 12 hours after entering MODE 2 from MODE 1.

The purpose of CTS Table 4.1.2 is to ensure the IRMs are OPERABLE when they are required to be OPERABLE. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. CTS Table 4.1.2 Note 1 requires the performance of an IRM calibration during every startup and normal shutdown. ITS SR 3.3.1.1.11 only requires the Surveillance to be performed every 24 months. The current testing requirements are excessive since they do not reflect the performance history of the IRMs. The IRMs will perform as designed as long as they are calibrated every 24 months, as is the case when the time between a startup and shutdown is the same as the refueling cycle interval. This calibration Frequency is consistent with the plant setpoint methodology. The purpose of ITS SR 3.3.1.1.11 Note 2 is to allow the plant to enter the Applicability of the Specification and allow the Surveillance to be performed 12 hours after entering MODE 2. This allowance is necessary to allow a normal shutdown to be completed and at the same time to allow time to perform the Surveillance. This change is necessary since testing of the IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. The proposed Surveillance Note provides a finite time in which the Surveillances must be performed after entering the specified condition and therefore this change is considered acceptable. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR. This portion of the change is considered administrative because the current frequency does not specify any time constraints. This change is less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.11 *(Category 7 – Relaxation Of Surveillance Frequency, Non-24 Month Type Change)* CTS 1.0.F requires the performance of a response time test once per cycle. ITS SR 3.3.1.1.14 requires the performance of a RPS RESPONSE TIME test every 24 months "on a STAGGERED TEST BASIS." ITS SR 3.3.1.1.14 is modified by a Note that states, "For Function 5, "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency." This changes the CTS by allowing the testing to be performed on a 24 month "STAGGERED TEST BASIS" instead of every 24 months. The change from once per cycle Frequency to a 24 month Frequency is discussed in DOC A.9.

The purpose of CTS 1.0.F is to ensure the response time testing is performed every 24 months. This change is acceptable because the new Surveillance Frequency ensures that it provides an acceptable level of equipment reliability. ITS SR 3.3.1.1.14 will allow this test to be performed every 24 months "on a STAGGERED TEST BASIS." The STAGGERED TEST BASIS definition has been added to the ITS Section 1.1 in accordance with the Discussion of Change in ITS Chapter 1.0. The STAGGERED TEST BASIS definition states "A STAGGERED TEST BASIS shall consist of the testing of one of the systems,

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subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function." ITS SR 3.3.1.1.14 is modified by a Note that states "For Function 5, "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency." The requirements in ITS SR 3.3.1.1.14 will require one channel of each Function associated with ITS 3.3.1.1-1 Functions 1.a, 2.a, 3, 4, 8, and 9 and "four" channels of Function 5 to be tested every 24 months. This change is acceptable since the RPS instrumentation is highly reliable. The Frequency is consistent with the industry and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. This change is less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.12 *(Category 10 – Changing Instrumentation Allowable Values)* CTS 3.1.A refers to the "setpoints" of the RPS Instrumentation Functions in CTS Table 3.1.1 and CTS Table 3.1.1 specifies the "Limiting Trip Settings" for the RPS Instrumentation Functions. The Limiting Trip Setting value of CTS Table 3.1.1 Trip Functions 3.a, 4.a, and 4.c have been modified to reflect new Allowable Values as indicated for ITS Table 3.3.1.1-1 Functions 1.a and 2.a. This changes the CTS by requiring the RPS Instrumentation to be set consistent with the new "Allowable Value." The change in the term "Limiting Trip Settings" to "Allowable Value" is discussed in DOC A.16.

The purpose of the Allowable Values is to ensure the instruments function as assumed in the safety analyses. ITS 3.3.1.1 reflects Allowable Values consistent with the philosophy of General Electric ISTS, NUREG-1433. These Allowable Values have been established using the GE setpoint methodology guidance, as specified in the Monticello setpoint methodology. The analytic limits are derived from limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits. The difference between the analytic limit and the Allowable Value allows for channel instrument accuracy, calibration accuracy, process measurement accuracy, and primary element accuracy. The margin between the Allowable Value and the nominal trip setpoint (NTSP) allows for instrument drift that might occur during the established surveillance period. Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift. These two verifications are statistical evaluations to provide additional assurance of the acceptability of the NTSP and may require changes to the NTSP. Use of these methods and verifications provides the assurance that if the setpoint is found conservative to the Allowable Value during surveillance testing, the instrumentation would have provided the required trip function by the time the process reached the analytic limit for the applicable events. Therefore, based on the above discussion, the inclusion of the Allowable Value as the OPERABILITY value in lieu of the Limiting Trip

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Setting is acceptable. This change is less restrictive because less stringent OPERABILITY values are being applied in the ITS than were applied in the CTS.

L.13 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS Table 4.1.2 requires the performance of an IRM calibration. ITS Table 3.3.1.1 Function 1 (IRM) requires the performance of SR 3.3.1.1.11, a CHANNEL CALIBRATION, however, the Surveillance includes a Note (Note 1) that excludes the neutron detectors from the calibration. This changes the CTS by not requiring the IRM neutron detectors to be tested.

The purpose of ITS SR 3.3.1.1.11 Note 1 is to exclude the neutron detectors from the IRM calibration. This change is acceptable because it has been determined that the relaxed Surveillance Requirement acceptance criteria are not necessary for verification that the equipment used to meet the LCO can perform its required functions. CTS Table 4.1.2 does not require the neutron detectors to be calibrated. A heat balance is the method of calibration required by CTS Table 4.1.2, and is performed and the IRM output is adjusted to conform to the heat balance. This test only adjusts the compensating voltage setting, not the actual neutron detector. This change is acceptable because the neutron detectors are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. This change is less restrictive because an explicit allowance has been included to exclude the IRM neutron detector from the calibration.

L.14 CTS Table 4.1.1 requires a weekly functional test of the Manual Scram Function. ITS Table 3.3.1.1-1 Function 11 and ITS SR 3.3.1.1.5 requires the performance of the same test at a 31 day Frequency. This changes the CTS by extending the Manual Scram functional test Frequency from 7 days to 31 days.

The purpose of the functional test is to ensure the Manual Scram Function instrumentation is functioning properly. This changes the CTS by extending the requirement to perform the test from 7 days to 31 days. The Manual Scram functional test Frequency was previously changed from monthly to weekly as part of the amendment request that adopted GE Topical Report NEDC-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," dated March 1988. NEDC-30851-P-A performed an analysis to extend the CHANNEL FUNCTIONAL TEST Frequency of the automatic RPS channels from monthly to quarterly. In order to justify this extension, it was necessary to actuate the automatic logic scram relays every 7 days. Therefore, NEDC-30851-P-A also changed the CHANNEL FUNCTIONAL TEST for the Manual Scram Function from monthly to weekly since, for the four manual scram pushbutton RPS design, performing a CHANNEL FUNCTIONAL TEST of the Manual Scram Functions (i.e., the manual pushbuttons) actuates the automatic logic scram relays. The Monticello amendment request adopting NEDC-30851-P-A included changing the Manual Scram functional test Frequency from monthly to weekly, and was approved by the NRC in License Amendment 81, dated April 16, 1992. However, the Manual Scram pushbuttons at Monticello do not actuate the automatic logic scram relays; a separate manual scram logic channel (designated A3 and B3) for each of the two manual scram pushbuttons is provided. Therefore, NEDC-30851-P-A did not actually require a change to the Manual Scram functional test frequency. To ensure the automatic logic scram

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relays are tested every week, the CTS Bases was updated on June 10, 2004 and clarifies that the Manual Scram refers to a manually initiated trip of both the Manual and Auto Scram logic. However, this change was made just to ensure the requirements of NEDC-30851-P-A, as they relate to the automatic logic scram relays, were met. ITS SR 3.3.1.1.4 has been included to ensure the automatic logic scram relays are tested every week. A review of past Manual Scram functional test Surveillances was performed and all completed tests were successful. Both monthly and weekly tests performed in 1992 (pre- and post-implementation of the monthly to weekly Surveillance Frequency change) and recent weekly tests were reviewed. In total, 27 completed Surveillances were reviewed and the Manual Scram functional test was successful in every case. Furthermore, the Manual Scram functional test only includes switches and relays and does not rely on instrument setpoints or other calibrations that are potentially subject to drift. Therefore, a monthly functional test Frequency for the Manual Scram and continued weekly testing of the automatic logic scram relays (to support NEDC-30851-P-A implementation) is acceptable. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.3 INSTRUMENTATION

3.1 3.3.1.1 Reactor Protection System (RPS) Instrumentation

3.1.A LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

3.1.A, Table 3.1.1 APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

-----NOTE-----

3.1.B Separate Condition entry is allowed for each channel.

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.1.B.1	A. One or more required channels inoperable.	A.1 Place channel in trip. OR A.2 Place associated trip system in trip.	12 hours 12 hours
3.1.B.2	B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours
3.1.B.2	C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour
3.1.B.3, Table 3.1.1 Note *	D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.1.B.3, Table 3.1.1 Required Condition D	E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < [30]% RTP. ≤ 45	4 hours	1
3.1.B.3, Table 3.1.1 Required Conditions B and C	F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2. INSERT 1	6 hours	11
3.1.B.3, Table 3.1.1 Required Condition A	G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 3.	12 hours	
3.1.B.3, Table 3.1.1 Required Condition A	H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately	

SURVEILLANCE REQUIREMENTS

-NOTES-

4.1.A 1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

Table 3.1.1 Note (1) 2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

CTS	SURVEILLANCE		FREQUENCY
Table 4.1.1 Note (2), DOC M.15	SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours

CTS

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INSERT 1

CTS		Required Action	Completion Time
		AND	
3.1.B.3, Table 3.1.1 Required Condition C	F.2	------------NOTE------------ Only applicable to Function 5. -------------------------------- Reduce reactor pressure to < 600 psig.	12 hours

CTS

SURVEILLANCE REQUIREMENTS (continued)

CTS	SURVEILLANCE		FREQUENCY	
Table 4.1.2, including Note 4	SR 3.3.1.1.2	------NOTE------ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. ------ Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP [plus ~~any~~ gain adjustment ~~required~~ by LCO 3.2.4, ~~"Average Power Range Monitor (APRM) Setpoints"~~] while operating at ≥ 25% RTP.	7 days	1
	~~SR 3.3.1.1.3~~	~~Adjust the channel to conform to a calibrated flow signal.~~	~~7 days~~	2
Table 4.1.1 Note 3	SR 3.3.1.1.~~4~~ 3	------NOTE------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ------ Perform CHANNEL FUNCTIONAL TEST.	7 days	2 3 INSERT 2
Table 4.1.1	SR 3.3.1.1.5	Perform CHANNEL FUNCTIONAL TEST.	~~7~~ 31 days	3
DOC M.14	SR 3.3.1.1.6	Calibrate the local power range monitors.	~~1000 MWD/T average core exposure~~ 2000 effective full power hours	5
Table 4.1.1	SR 3.3.1.1.7	Perform CHANNEL FUNCTIONAL TEST.	[92] days	1
Table 4.1.2	SR 3.3.1.1.8	[Calibrate the trip units.	[92] days]	1

CTS

3

INSERT 2

DOC M.10	SR 3.3.1.1.4	Perform a functional test of each RPS automatic scram contactor.	7 days

CTS

SURVEILLANCE REQUIREMENTS (continued)

CTS		SURVEILLANCE	FREQUENCY	
Table 4.1.2, DOC M.14	SR 3.3.1.1.9	---NOTE~~S~~--- ~~1.~~ Neutron detectors are excluded. ~~2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.~~ --- Perform CHANNEL CALIBRATION.	~~[184]~~ 92 days	8 4
Table 4.1.1	SR 3.3.1.1.10	Perform CHANNEL FUNCTIONAL TEST.	~~[18]~~ 24 months	1
Table 4.1.2, including Note 1	SR 3.3.1.1.11	---NOTES--- 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL CALIBRATION.	~~[18]~~ 24 months	1
	~~SR 3.3.1.1.12~~	~~Verify the APRM Flow Biased Simulated Thermal Power - High time constant is ≤ [7] seconds.~~	[18] months	6
DOC M.1	SR 3.3.1.1.~~13~~ 12	Perform LOGIC SYSTEM FUNCTIONAL TEST.	~~[18]~~ 24 months	6 1
DOC M.11	SR 3.3.1.1.~~14~~ 13	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, ~~Trip~~ Oil Pressure - Low Acceleration Relay Functions are not bypassed when THERMAL POWER is ~~≥ [30]~~ > 45% RTP.	~~[18]~~ 24 months	6 1 7 1

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY	
	SR 3.3.1.1.~~15~~ 14 -------------------------NOTE~~S~~-------------------------		6
3.1.A, 1.0.F	~~1 Neutron detectors are excluded.~~		9
	~~2.~~ For Function 5 "n" equals 4 channels for the purpose of determining the ~~the~~ STAGGERED TEST BASIS Frequency.		10

	Verify the RPS RESPONSE TIME is within limits.	24 ~~[18]~~ months on a STAGGERED TEST BASIS	1

Table 3.3.1.1-1 (page 1 of 4)
Reactor Protection System Instrumentation

CTS

Table and Trip Function or Instrument Channel Number	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	1 REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	2 3 4 6 13 14 SURVEILLANCE REQUIREMENTS	1 ALLOWABLE VALUE
	1. Intermediate Range Monitors			SR 3.3.1.1.4		
3.1.1 (3.a), 4.1.1 (11), 4.1.2 (2)	a. Neutron Flux - High (High 7)	2	[3]	G	SR 3.3.1.1.1 (3) SR 3.3.1.1.~~4~~ SR 3.3.1.1.11 (12) SR 3.3.1.1.~~13~~ SR 3.3.1.1.14	≤ [~~120~~/125] 122 divisions of full scale
		5[(a)]	[3]	H	SR 3.3.1.1.1 (3) SR 3.3.1.1.~~5~~ SR 3.3.1.1.11 (12) SR 3.3.1.1.~~13~~ SR 3.3.1.1.14	≤ [~~120~~/125] 122 divisions of full scale
3.1.1 (3.b), 4.1.1 (11), 4.1.2 (2)	b. Inop	2	[3]	G	SR 3.3.1.1.~~4~~ (3) SR 3.3.1.1.~~13~~ (12)	NA
		5[(a)]	[3]	H	SR 3.3.1.1.~~5~~ (3) SR 3.3.1.1.~~13~~ (12)	NA
	2. Average Power Range Monitors					
	~~a. Neutron Flux - High, Setdown~~ (8)	~~2~~	~~[2]~~	~~G~~	~~SR 3.3.1.1.1~~ ~~SR 3.3.1.1.4~~ ~~SR 3.3.1.1.6~~ ~~SR 3.3.1.1.9~~ ~~SR 3.3.1.1.13~~	~~≤ [20]% RTP~~
3.1.1 (4.a) and (4.c), 4.1.1 (10), 4.1.2 (1, 11)	~~b.~~ a (8). Flow ~~Biased~~ Referenced Neutron Flux ~~Simulated Thermal Power~~ - High (High 6)	1	[2]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 ~~SR 3.3.1.1.3~~ SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.9 ~~SR 3.3.1.1.12~~ SR 3.3.1.1.~~13~~ (12) SR 3.3.1.1.~~15~~ (14)	≤ [~~0.58~~] 0.66 W + [~~62~~] 67.6 % RTP and ≤ [1~~15.5~~] 122 % RTP[(b)]

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

3.1.1 (4.a) (b) [0.58 W + ~~62%~~ - 0.58 ΔW] 0.66 (W - 5.4) + 67.6% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." (1)

Table 3.3.1.1-1 (page 2 of 4)
Reactor Protection System Instrumentation

CTS Table and Trip Function or Instrument Channel Number	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM (1)	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS (2) (3) (4) (6) (12)	ALLOWABLE VALUE (1)
	2. Average Power Range Monitors					
	c. Fixed Neutron Flux - High	1	[2]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ [120]% RTP
	[d. Downscale	1	[2]	F	SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13	≥ [3]% RTP]
3.1.1 (4.b), 4.1.1 (10), 4.1.2 (1)	e. b. Inop (8)	1,2	[2]	G F	SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 12	NA
3.1.1 (5), 4.1.1 (1), 4.1.2 (3)	3. Reactor Vessel Steam Dome Pressure - High	1,2	[2]	G	SR 3.3.1.1.1 SR 3.3.1.1.7 [SR 3.3.1.1.8] 9 SR 3.3.1.1.11 12 SR 3.3.1.1.13 14 SR 3.3.1.1.15	≤ [1054] 1075 psig
3.1.1 (7), 4.1.1 (3), 4.1.2 (5)	4. Reactor Vessel Water Level - Low, Level 3 (7)	1,2	[2]	G	SR 3.3.1.1.1 SR 3.3.1.1.7 [SR 3.3.1.1.8] SR 3.3.1.1.11 12 SR 3.3.1.1.13 14 SR 3.3.1.1.15	≥ [10] 7 inches
3.1.1(10), 4.1.1 (6), 4.1.2 (8)	5. Main Steam Isolation Valve - Closure	1,2[(c)] (11)	[8]	F	SR 3.3.1.1.7 SR 3.3.1.1.11 12 SR 3.3.1.1.13 14 SR 3.3.1.1.15	≤ [10]% closed
3.1.1 (6), 4.1.1 (2), 4.1.2 (4)	6. Drywell Pressure - High	1,2	[2]	G	SR 3.3.1.1.1 SR 3.3.1.1.7 [SR 3.3.1.1.8] 9 SR 3.3.1.1.11 12 SR 3.3.1.1.13 SR 3.3.1.1.4	≤ [1.92] 2.0 psig

(c) With reactor pressure ≥ 600 psig (11)

Table 3.3.1.1-1 (page 3 of 4)
Reactor Protection System Instrumentation

CTS

Table and Trip Function or Instrument Channel Number	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	1 REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	2 3 4 6 12 SURVEILLANCE REQUIREMENTS	1 ALLOWABLE VALUE
	7. Scram Discharge Volume Water Level - High			SR 3.3.1.1.4		
3.1.1 (8), 4.1.1 (4), 4.1.2 (6)	a. Resistance Temperature Detector	1,2	[2]	G	[SR 3.3.1.1.1] SR 3.3.1.1.7 [SR 3.3.1.1.8] 9 SR 3.3.1.1.[11] 12 SR 3.3.1.1.[13]	≤ [57.15] 56.0 gallons
		5(a)	[2]	H	[SR 3.3.1.1.1] SR 3.3.1.1.7 [SR 3.3.1.1.8] 9 SR 3.3.1.1.[11] 12 SR 3.3.1.1.[13]	≤ [57.15] 56.0 gallons
3.1.1 (8), 4.1.1 (4), 4.1.2 (6)	b. Float Switch	1,2	[2]	G	SR 3.3.1.1.7 9 SR 3.3.1.1.[11] SR 3.3.1.1.[13] 12	≤ [57.15] 56.0 gallons
		5(a)	[2]	H	SR 3.3.1.1.7 9 SR 3.3.1.1.[11] SR 3.3.1.1.[13] 12	≤ [57.15] 56.0 gallons
3.1.1 (12), 4.1.1 (7), 4.1.2 (10)	8. Turbine Stop Valve - Closure	45 1 ≥ [30]% RTP	[4]	E	SR 3.3.1.1.7 [SR 3.3.1.1.8] SR 3.3.1.1.11 SR 3.3.1.1.[13] 12 SR 3.3.1.1.[14] 13 SR 3.3.1.1.[15] 14	≤ [10]% closed
3.1.1 (11), 4.1.1 (9), 4.1.2 (9)	9. Turbine Control Valve Fast Closure, [Trip] Oil Pressure - Low Acceleration Relay 7	45 1 ≥ [30]% RTP	[2]	E	SR 3.3.1.1.7 [SR 3.3.1.1.8] SR 3.3.1.1.[11] 9 SR 3.3.1.1.[13] 12 SR 3.3.1.1.[14] 13 SR 3.3.1.1.[15] 14	≥ [600] 167.8 psig
3.1.1 (1), 4.1.1 (12)	10. Reactor Mode Switch - Shutdown Position	1,2	[2] 1	G	SR 3.3.1.1.10 SR 3.3.1.1.[13] 12	NA
		5(a)	[2] 1	H	SR 3.3.1.1.10 SR 3.3.1.1.[13] 12	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.1.1-1 (page 4 of 4)
Reactor Protection System Instrumentation

CTS

Table and Trip Function or Instrument Channel Number	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	1 REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	2 6 SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3.1.1 (2), 4.1.1 (8)	11. Manual Scram	1,2	[2] 1	G	SR 3.3.1.1.5 SR 3.3.1.1.[13] 12	NA
		5(a)	[2] 1	H	SR 3.3.1.1.5 SR 3.3.1.1.[13] 12	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.1.1, REACTOR PROTECTION SYSTEM (RPS) INSTRUMENATION

1. The brackets are removed and the proper plant specific information/value is provided.

2. The Frequency of ISTS SR 3.3.1.1.3 (ITS SR 3.3.1.1.9) has been changed from 7 days to 92 days as part of a CHANNEL CALIBRATION, consistent with the current licensing basis. Subsequent SRs have been renumbered, as necessary.

3. ITS SR 3.3.1.1.4 has been added to require the performance of a functional test of each RPS automatic scram contactor. Each RPS trip system at Monticello includes three independent channels, two automatic and one manual (A1, A2, and A3 for one trip system and B1, B2, and B3 for the other trip system), while the standard BWR RPS design includes only two channels (A1 and A2 for one trip system and B1 and B2 for the other trip system), and the Manual Scram Function inputs directly into both of the channels of a trip system. This SR has been added to functionally test each RPS automatic scram contractor. This functional test has been added to allow the CHANNEL FUNCTIONAL TEST Surveillance test intervals to be extended as justified in NEDC-30851-P-A and approved for Monticello in Amendment 81 (dated April 16, 1992) since the Monticello RPS logic design is different from the generic BWR RPS design. Since the contactors are required for the OPERABILITY of each automatic Function, the Surveillance has been associated with each automatic scram Function in Table 3.3.1.1-1. The Manual Scram Function (ITS Table 3.3.1.1-1 Function 11) will be tested every 31 days as shown in ITS SR 3.3.1.1.5. Subsequent SRs have been renumbered as required.

4. The ISTS SR 3.3.1.1.9 Frequency has been changed from 184 days to 92 days for ITS Table 3.3.1.1-1 Functions 2.a, 3, 6, 7.a, 7.b, and 9. This is the current licensing basis calibration frequency for these RPS Instrumentation Functions.

5. The Frequency for ISTS SR 3.3.1.1.6 has been changed from 1000 MWD/T to 2000 effective full power hours. This Surveillance Frequency is consistent with current plant practice for calibration of the local power range monitors.

6. ISTS SR 3.3.1.1.12 has been deleted since the APRM circuitry design does not include a time constant. ISTS Table 3.3.1.1-1 Function 2.b (ITS Table 3.3.1.1-1 Function 2.a) has been renamed consistent with current nomenclature. Subsequent SRs have been renumbered as necessary.

7. Changes are made (additions, deletions, and/or changes) to the ISTS, which reflect the plant specific nomenclature.

8. ISTS Table 3.3.1.1-1 Function 2.a (Neutron Flux - High, Setdown), Function 2.c (Fixed Neutron Flux - High), and Function 2.d (Downscale) do not exist in the Monticello design. Therefore, these Functions are deleted and subsequent Functions have been renumbered, as applicable. The Applicability of ISTS Table 3.3.1.1-1 Function 2.e (the APRM Inop Function) has been changed from MODES 1 and 2 to MODE 1 to be consistent with the MODE proposed for ITS 3.3.1.1-1 Function 2.a (Average Power Range Monitors Flow Referenced Neutron Flux - High High). Similarly the Condition referenced in the fourth column of the Table has been changed from G to F, since ACTION F requires the plant to be in MODE 2, which is outside of the Applicability of ITS Table 3.3.1.1-1 Function 2.b.

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.1.1, REACTOR PROTECTION SYSTEM (RPS) INSTRUMENATION

Also, Note 2 to ISTS SR 3.3.1.1.9 has been deleted since Function 2.a has been deleted.

9. This Note in ISTS SR 3.3.1.1.15 (ITS SR 3.3.1.1.14) has been deleted since the RPS RESPONSE TIME definition has been changed to be consistent with the current licensing basis. The RPS RESPONSE TIME definition excludes the requirement to test sensor response time. Therefore, the allowance in the ISTS Note is not needed because it is already included in the definition. The subsequent Note numbering has been deleted since there is only one remaining Note.

10. Grammatical error corrected.

11. ITS Required Action F.2 for Function 5, Main Steam Isolation Valve - Closure, has been added to require reducing reactor pressure to < 600 psig. The Applicability of ITS Table 3.3.1.1-1 Function 5 has also been revised to include MODE 2 and footnote (c), i.e., MODE 2 with reactor pressure $\geq$ 600 psig. These changes are consistent with the current licensing basis for the Main Steam Isolation Valve – Closure Function.

12. The CHANNEL CHECK Surveillance in ISTS Table 3.3.1.1-1 associated with the Reactor Vessel Steam Dome Pressure - High, Drywell Pressure - High, Scram Discharge Volume Water Level - High (Resistance Temperature Detector) Functions has been deleted because the associated channels at Monticello include a switch or thermal probe and there is no available method to verify channel indication.

13. An RPS RESPONSE TIME test (ITS SR 3.3.1.1.14) has been added for ITS Table 3.3.1.1-1 Function 1.a, IRM Neutron Flux - High High Function. This has been added since the accident analysis directly credits this Function.

14. The CHANNEL FUNCTIONAL TEST associated with the IRMs in MODE 5 (ISTS SR 3.3.1.1.5) has been renumbered as SR 3.3.1.1.3, since ISTS SR 3.3.1.1.4 (ITS SR 3.3.1.1.3) is the CHANNEL FUNCTIONAL TEST for this Function in MODE 2. Therefore, for consistency, a single SR is specified for the CHANNEL FUNCTIONAL TEST requirement in all required MODES. In addition, the ISTS Bases for ISTS SR 3.3.1.1.5 implies that the ISTS SR 3.3.1.1.5 CHANNEL FUNCTIONAL TEST is only for the Manual Scram Function.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION

B 3.3.1.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor scram when one or more monitored parameters exceed their specified limits, to preserve the integrity of the fuel cladding and the Reactor Coolant System (RCS) and minimize the energy that must be absorbed following a loss of coolant accident (LOCA). This can be accomplished either automatically or manually.

The protection and monitoring functions of the RPS have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance. Technical Specifications are required by 10 CFR 50.36 to contain LSSS defined by the regulation as "...settings for automatic protective devices...so chosen that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytic Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytic Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protective devices must be chosen to be more conservative than the Analytic Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

The trip setpoint is a predetermined setting for a protective device chosen to ensure automatic actuation prior to the process variable reaching the Analytic Limit and thus ensuring that the SL would not be exceeded. As such, the trip setpoint accounts for uncertainties in setting the device (e.g., calibration), uncertainties in how the device might actually perform (e.g., repeatability), changes in the point of action of the device over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the trip setpoint plays an important role in ensuring that SLs are not exceeded. As such, the trip setpoint meets the definition of an LSSS (Ref. 1) and could be used to meet the requirement that they be contained in the Technical Specifications.

BASES

BACKGROUND (continued)

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. Operable is defined in Technical Specifications as "...being capable of performing its safety function(s)." For automatic protective devices, the required safety function is to ensure that a SL is not exceeded and therefore the LSSS as defined by 10 CFR 50.36 is the same as the OPERABILITY limit for these devices. However, use of the trip setpoint to define OPERABILITY in Technical Specifications and its corresponding designation as the LSSS required by 10 CFR 50.36 would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as found" value of a protective device setting during a Surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protective device with a setting that has been found to be different from the trip setpoint due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the trip setpoint and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as found" setting of the protective device. Therefore, the device would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the device to the trip setpoint to account for further drift during the next surveillance interval. 1

Use of the trip setpoint to define "as found" OPERABILITY and its designation as the LSSS under the expected circumstances described above would result in actions required by both the rule and Technical Specifications that are clearly not warranted. However, there is also some point beyond which the device would have not been able to perform its function due, for example, to greater than expected drift. This value needs to be specified in the Technical Specifications in order to define OPERABILITY of the devices and is designated as the Allowable Value which, as stated above, is the same as the LSSS.

The Allowable Valuable specified in Table 3.3.1-1 serves as the LSSS such that a channel is OPERABLE if the trip setpoint is found not to exceed the Allowable Value. As such, the Allowable Value differs from the trip setpoint by an amount primarily equal to the expected instrument loop uncertainties, such as drift, during the surveillance interval. In this

All changes are "2" unless otherwise noted

BASES

BACKGROUND (continued)

manner, the actual setting of the device will still meet the LSSS definition and ensure that a SL is not exceeded at any given point of time as long as the device has not drifted beyond that expected during the surveillance interval. If the actual setting of the device is found to have exceeded the Allowable Value the device would be considered inoperable from a Technical Specification perspective. This requires corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required. Note that, although the channel is "OPERABLE" under these circumstances, the trip setpoint should be left adjusted to a value within the established trip setpoint calibration tolerance band, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned.

described U Section 7.6.1.2.1 3

The RPS, as ~~shown~~ in ~~the~~ ~~F~~SAR, ~~Figure [ ]~~ (Ref. 2), includes sensors, relays, bypass circuits, and switches that are necessary to cause initiation of a reactor scram. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. The input parameters to the scram logic are from instrumentation that monitors reactor vessel water level, reactor vessel pressure, neutron flux, main steam line isolation valve position, turbine control valve (TCV) ~~fast closure~~ ~~trip~~ oil pressure, turbine stop valve (TSV) position, drywell pressure, and scram discharge volume (SDV) water level, as well as reactor mode switch in shutdown position and manual scram signals. There are at least four redundant sensor input signals from each of these parameters (with the exception of the reactor mode switch in shutdown scram signal). ~~Most~~ channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an RPS trip signal to the trip logic. ~~Table B 3.3.1.1-1 summarizes the diversity of sensors capable of initiating scrams during anticipated operating transients typically analyzed.~~

acceleration relay

and manual

s

Some

4

automatic , and A3, three

The RPS is comprised of two independent trip systems (A and B) with ~~two~~ logic channels in each trip system (logic channels A1 ~~and~~ A2, B1 ~~and~~ B2), as ~~shown~~ in Reference ~~2~~. The outputs of the logic channels in a trip system are combined in a one-out-of-two logic so that either channel can trip the associated trip system. The tripping of both trip systems will produce a reactor scram. This logic arrangement is referred to as a one-out-of-two taken twice logic. Each trip system can be reset by use of a reset switch. If a full scram occurs (both trip systems trip), a relay prevents reset of the trip systems for ~~10 seconds~~ after the full scram signal is received. This ~~10 second~~ delay on reset ensures that the scram function will be completed.

described , and B3

3 , INSERT 1

INSERT 2

a short time delay

e short time

2

INSERT 1

The automatic trip logics of trip system A are logic channels A1 and A2; the manual trip logic of trip system A is logic channel A3. Similarly, the trip logics for trip system B are logic channels B1, B2, and B3.

2

INSERT 2

The outputs of the manual logic channels in a trip system are combined in a one-out-of-one logic. The tripping of both manual logic channels will produce a scram.

BASES

BACKGROUND (continued)

Two scram pilot valves are located in the hydraulic control unit for each control rod drive (CRD). Each scram pilot valve is solenoid operated, with the solenoids normally energized. The scram pilot valves control the air supply to the scram inlet and outlet valves for the associated CRD. When either scram pilot valve solenoid is energized, air pressure holds the scram valves closed and, therefore, both scram pilot valve solenoids must be de-energized to cause a control rod to scram. The scram valves control the supply and discharge paths for the CRD water during a scram. One of the scram pilot valve solenoids for each CRD is controlled by trip system A, and the other solenoid is controlled by trip system B. Any trip of trip system A in conjunction with any trip in trip system B results in de-energizing both solenoids, air bleeding off, scram valves opening, and control rod scram.

The backup scram valves, which energize on a scram signal to depressurize the scram air header, are also controlled by the RPS. Additionally, the RPS System controls the SDV vent and drain valves such that when both trip systems trip, the SDV vent and drain valves close to isolate the SDV.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The actions of the RPS are assumed in the safety analyses of References 4 and 5 2, 3, ~~and~~ 4. The RPS initiates a reactor scram when monitored parameter values exceed the Allowable Values, specified by the setpoint methodology and listed in Table 3.3.1.1-1 to preserve the integrity of the fuel cladding, the reactor coolant pressure boundary (RCPB), and the containment by minimizing the energy that must be absorbed following a LOCA. 2

RPS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Functions not specifically credited in the accident analysis are retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The OPERABILITY of the RPS is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.1.1-1. Each Function must have a required number of OPERABLE channels per RPS trip system, with their setpoints within the specified Allowable Value, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Each channel must also respond within its assumed response time.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

most of the   s   6

Allowable Values are specified for each RPS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

2

INSERT 2A

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. ~~The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.~~

The OPERABILITY of scram pilot valves and associated solenoids, backup scram valves, and SDV valves, described in the Background section, are not addressed by this LCO.

The individual Functions are required to be OPERABLE in the MODES specified in the table, which may require an RPS trip to mitigate the consequences of a design basis accident or transient. To ensure a reliable scram function, a combination of Functions are required in each MODE to provide primary and diverse initiation signals.

MODES 1 and 2, and

In MODE 5,

The RPS is required to be OPERABLE in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. 5
Control rods withdrawn from a core cell containing no fuel assemblies do not affect the reactivity of the core and, therefore, are not required to have the capability to scram. Provided all other control rods remain inserted, the RPS function is not required. In this condition, the required SDM (LCO 3.1.1) and refuel position one-rod-out interlock (LCO 3.9.2) ensure that no event requiring RPS will occur. During normal operation in 5

,"SHUTDOWN MARGIN"

, "Refueling Position One-Rod-Out Interlock"

Include remainder of sentence on next page

8

2

INSERT 2A

The Allowable Values and nominal trip setpoints (NTSP) are derived, using the General Electric setpoint methodology guidance, as specified in the Monticello setpoint methodology. The Allowable Values are derived from the analytic limits. The difference between the analytic limit and the Allowable Value allows for channel instrument accuracy, calibration accuracy, process measurement accuracy, and primary element accuracy. The margin between the Allowable Value and the NTSP allows for instrument drift that might occur during the established surveillance period. Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift. These two verifications are statistical evaluations to provide additional assurance of the acceptability of the NTSP and may require changes to the NTSP. Use of these methods and verifications provides the assurance that if the setpoint is found conservative to the Allowable Value during surveillance testing, the instrumentation would have provided the required trip function by the time the process reached the analytic limit for the applicable events.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued) 8

, "Control Rod Block Instrumentation"

MODES 3 and 4, all control rods are fully inserted and the Reactor Mode Switch Shutdown Position control rod withdrawal block (LCO 3.3.2.1) does not allow any control rod to be withdrawn. Under these conditions, the RPS function is not required to be OPERABLE. 5

move to page B 3.3.1.1-5 as indicated

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Intermediate Range Monitor (IRM)

High

1.a. Intermediate Range Monitor Neutron Flux - High 6

The IRMs monitor neutron flux levels from the upper range of the source range monitor (SRM) to the lower range of the average power range monitors (APRMs). The IRMs are capable of generating trip signals that can be used to prevent fuel damage resulting from abnormal operating transients in the intermediate power range. In this power range, the most significant source of reactivity change is due to control rod withdrawal. The IRM provides diverse protection for the rod worth minimizer (RWM), which monitors and controls the movement of control rods at low power. The RWM prevents the withdrawal of an out of sequence control rod during startup that could result in an unacceptable neutron flux excursion (Ref. 6 ~~3~~). The IRM provides mitigation of the neutron flux excursion. To demonstrate the capability of the IRM System to mitigate control rod withdrawal events, generic analyses have been performed (Ref. 7 ~~4~~) to evaluate the consequences of control rod withdrawal events during startup that are mitigated only by the IRM. This analysis, which assumes that one IRM channel in each trip system is bypassed, demonstrates that the IRMs provide protection against local control rod withdrawal errors and results in peak fuel energy depositions below the 170 cal/gm fuel failure threshold criterion. 2 2

The IRMs are ~~also~~ capable of limiting other reactivity excursions during startup, such as cold water injection events, although no credit is specifically assumed. 2 2

INSERT 2B

7 2

The IRM System is divided into two groups of IRM channels, with four IRM channels inputting to each trip system. The analysis of Reference ~~4~~ assumes that one channel in each trip system is bypassed. Therefore, six channels with three channels in each trip system are required for IRM OPERABILITY to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This trip is active in each of the 10 or 12 ranges of the IRM, which must be selected by the operator to maintain the neutron flux within the monitored level of an IRM range.

For an IRM to be considered OPERABLE it must be fully inserted. 2

2 **INSERT 2B**

This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued) 2

7

The analysis of Reference $\boxed{4}$ has adequate conservatism to permit an IRM Allowable Value of $\boxed{120}$ divisions of a 125 division scale. INSERT 2C 2

121.5 High

The Intermediate Range Monitor Neutron Flux - High Function must be OPERABLE during MODE 2 when control rods may be withdrawn and the potential for criticality exists. In MODE 5, when a cell with fuel has its control rod withdrawn, the IRMs provide monitoring for and protection against unexpected reactivity excursions. In MODE 1, the APRM System and the RWM provide protection against control rod withdrawal error events and the IRMs are not required. 6

1.b. Intermediate Range Monitor – Inop

This trip signal provides assurance that a minimum number of IRMs are OPERABLE. Anytime an IRM mode switch is moved to any position other than "Operate," the detector voltage drops below a preset level, or when a module is not plugged in, an inoperative trip signal will be received by the RPS unless the IRM is bypassed. Since only one IRM in each trip system may be bypassed, only one IRM in each RPS trip system may be inoperable without resulting in an RPS trip signal.

This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

Six channels of Intermediate Range Monitor - Inop with three channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

Since this Function is not assumed in the safety analysis, there is no Allowable Value for this Function.

This Function is required to be OPERABLE when the Intermediate Range Monitor Neutron Flux - High Function is required. 6

High

2 **INSERT 2C**

In addition, the Allowable Value ensures that a reactor scram occurs before reaching 20% RTP on the highest IRM range. This indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Average Power Range Monitor

Flow Referenced High

2.a. Average Power Range Monitor Neutron Flux - High, Setdown (6)

The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP. For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux - High, Setdown Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High, Setdown Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High, Setdown Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High, Setdown Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP. (2)

The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Neutron Flux - High, Setdown with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 11 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

All changes are "2" unless otherwise noted

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~The Average Power Range Monitor Neutron Flux - High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists. In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.~~

~~2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power - High~~

High

Referenced Neutron Flux

6

The Average Power Range Monitor Flow ~~Biased Simulated Thermal Power~~ - High Function monitors neutron flux ~~to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor~~. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit ~~that is always lower than the Average Power Range Monitor Fixed Neutron Flux - High Function Allowable Value~~. The Average Power Range Monitor Flow ~~Biased Simulated Thermal Power~~ - High Function provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event) and protects the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. ~~During these events, the THERMAL POWER increase does not significantly lag the neutron flux response and, because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Fixed Neutron Flux - High Function will provide a scram signal before the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function setpoint is exceeded.~~

Referenced Neutron Flux

High

6

INSERT 3

three

The APRM System is divided into two groups of channels with ~~four~~ APRM inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Flow ~~Biased Simulated Thermal Power~~ - High, with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In

channel

Referenced Neutron Flux

High

6

6

2

INSERT 3

During any transient event that occurs at a reduced recirculation flow, because of a lower scram trip setpoint, the Average Power Range Monitor Flow Referenced Neutron Flux - High High Function will initiate a scram before the clamped Allowable Value is reached. However, the clamped value is not credited in the transient analyses except for the plant stability analysis (Ref. 8). The Average Power Range Monitor Flow Referenced Neutron Flux - High High Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 9 the Average Power Range Monitor Flow Referenced Neutron Flux - High High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 10) takes credit for the Average Power Range Monitor Flow Referenced Neutron Flux - High High Function to terminate the CRDA.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2 INSERT 4

2 INSERT 3A 14

addition, to provide adequate coverage of the entire core, at least ~~11~~ LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located. ~~Each APRM channel receives two total drive flow signals representative of total core flow. The total drive flow signals are generated by four flow units, two of which supply signals to the trip system A APRMs, while the other two supply signals to the trip system B APRMs. Each flow unit signal is provided by summing up the flow signals from the two recirculation loops. To obtain the most conservative reference signals, the total flow signals from the two flow units (associated with a trip system as described above) are routed to a low auction circuit associated with each APRM. Each APRM's auction circuit selects the lower of the two flow unit signals for use as the scram trip reference for that particular APRM. Each required Average Power Range Monitor Flow Biased Simulated Thermal Power - High channel only requires an input from one OPERABLE flow unit, since the individual APRM channel will perform the intended function with only one OPERABLE flow unit input. However, in order to maintain single failure criteria for the Function, at least one required Average Power Range Monitor Flow Biased Simulated Thermal Power - High channel in each trip system must be capable of maintaining an OPERABLE flow unit signal in the event of a failure of an auction circuit, or a flow unit, in the associated trip system (e.g., if a flow unit is inoperable, one of the two required Average Power Range Monitor Flow Biased Simulated Thermal Power - High channels in the associated trip system must be considered inoperable).~~

2 INSERT 5

~~The clamped Allowable Value is based on analyses that take credit for the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function for the mitigation of the loss of feedwater heating event. The THERMAL POWER time constant of < 7 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER.~~

High Referenced Neutron Flux 6

The Average Power Range Monitor Flow ~~Biased Simulated Thermal Power~~ - High Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). ~~During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.~~

INSERT 6 2

2

INSERT 3A

except that APRM channels 1, 2, 5, and 6 may lose all LPRM inputs from the companion APRM cabinet plus one additional LPRM input

2

INSERT 4

Each APRM channel receives one total drive flow signal representative of total core flow. The total drive flow signals are generated by two flow converters, one of which supplies signals to the trip system A APRMs, while the other supplies signals to the trip system B APRMs. Each flow converter signal is provided by summing up a flow signal from the two recirculation loops. Each required Average Power Range Monitor Flow Referenced Neutron Flux - High High channel requires an input from one OPERABLE flow converter (e.g., if a converter unit is inoperable, the associated Average Power Range Monitor Flow Referenced Neutron Flux - High High channels must be considered inoperable). An APRM flow converter is considered inoperable whenever it cannot deliver a flow signal less than or equal to actual recirculation flow conditions for all steady state and transient reactor conditions while in MODE 1. Reduced flow or downscale flow converter conditions due to planned maintenance or testing activities during derated unit conditions (i.e., end of cycle coast down) will result in conservative setpoints for the APRM Flow Referenced Neutron Flux - High High Function, thus maintaining the Function OPERABLE.

2

INSERT 5

The Allowable Value is selected to ensure the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. "W," in the Allowable Value column of Table 3.3.1.1-1, is the percentage of recirculation loop flow that provides a rated core flow of 57.6 million lbs/hr. The clamped Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

2

INSERT 6

Although the Average Power Range Monitor Flow Referenced Neutron Flux - High High Function is assumed in the CRDA analysis, which is applicable in MODE 2, the IRM Neutron Flux - High High Function conservatively bounds the assumed trip and provides adequate protection. Therefore, the Average Power Range Monitor Flow Referenced Neutron Flux - High High Function is not required in MODE 2.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.c. Average Power Range Monitor Fixed Neutron Flux - High

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The Average Power Range Monitor Fixed Neutron Flux - High Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 5, the Average Power Range Monitor Fixed Neutron Flux - High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 6) takes credit for the Average Power Range Monitor Fixed Neutron Flux - High Function to terminate the CRDA.

The APRM System is divided into two groups of channels with three APRM channels inputting to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Fixed Neutron Flux - High with two channels in each trip system arranged in a one-out-of- two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 11 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located. 6

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Fixed Neutron Flux - High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Fixed Neutron Flux - High Function is assumed in the CRDA analysis, which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux - High, Setdown Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor Fixed Neutron Flux - High Function is not required in MODE 2.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~2.d. Average Power Range Monitor – Downscale~~

~~This signal ensures that there is adequate Neutron Monitoring System protection if the reactor mode switch is placed in the run position prior to the APRMs coming on scale. With the reactor mode switch in run, an APRM downscale signal coincident with an associated Intermediate Range Monitor Neutron Flux - High or Inop signal generates a trip signal. This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.~~ 6

~~The APRM System is divided into two groups of channels with three inputs into each trip system. The system is designed to allow one channel in each trip system to be bypassed. Four channels of Average Power Range Monitor - Downscale with two channels in each trip system arranged in a one-out-of- two logic are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. The Intermediate Range Monitor Neutron Flux - High and Inop Functions are also part of the OPERABILITY of the Average Power Range Monitor - Downscale Function (i.e., if either of these IRM Functions cannot send a signal to the Average Power Range Monitor - Downscale Function, the associated Average Power Range Monitor - Downscale channel is considered inoperable).~~

~~The Allowable Value is based upon ensuring that the APRMs are in the linear scale range when transfers are made between APRMs and IRMs.~~

~~This Function is required to be OPERABLE in MODE 1 since this is when the APRMs are the primary indicators of reactor power.~~

d

2.~~e~~. Average Power Range Monitor – Inop 6

This signal provides assurance that a minimum number of APRMs are OPERABLE. Anytime an APRM mode switch is moved to any position 19 other than "Operate," an APRM module is unplugged, ~~the electronic operating voltage is low,~~ or the APRM has too few LPRM inputs (< ~~11~~), an inoperative trip signal will be received by the RPS, unless the APRM is bypassed. Since only one APRM in each trip system may be bypassed, only one APRM in each trip system may be inoperable without resulting in an RPS trip signal. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis. 2 6

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Four channels of Average Power Range Monitor - Inop with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal.

There is no Allowable Value for this Function.

This Function is required to be OPERABLE in the MODES where the APRM Functions are required.

3. Reactor Vessel Steam Dome Pressure - High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. No specific safety analysis takes direct credit for this Function. However, the Reactor Vessel Steam Dome Pressure - High Function initiates a scram for transients that results in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 5, reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux - High signal, not the Reactor Vessel Steam Dome Pressure - High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

9 2 Flow Referenced High 6 2 switches

High reactor pressure signals are initiated from four pressure ~~transmitters~~ that sense reactor pressure. The Reactor Vessel Steam Dome Pressure - High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure - High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required to be OPERABLE in MODES 1 and 2 when the RCS is pressurized and the potential for pressure increase exists.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4. Reactor Vessel Water Level - Low, ~~Level~~ 3 6

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, a reactor scram is initiated at ~~Level 3~~ this level to substantially reduce the heat generated in the fuel from fission. The Reactor Vessel Water Level - Low, ~~Level~~ 3 Function is not assumed in the analysis of the recirculation line break (Ref. ~~7~~). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the Emergency Core Cooling Systems (ECCS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. 2 6

2 INSERT 7 11

Reactor Vessel Water Level - Low, ~~Level~~ 3 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. 6

Four channels of Reactor Vessel Water Level - Low, ~~Level~~ 3 Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. 6

The Reactor Vessel Water Level - Low, ~~Level~~ 3 Allowable Value is selected to ensure that during normal operation the separator skirts are not uncovered (this protects available recirculation pump net positive suction head (NPSH) from significant carryunder) and for transients involving loss of all normal feedwater flow, initiation of the low pressure ECCS subsystems at Reactor Vessel Water Level - Low Low Low, ~~Level~~ 1 will not be required. 6 2

The Function is required in MODES 1 and 2 where considerable energy exists in the RCS resulting in the limiting transients and accidents. ECCS initiations at Reactor Vessel Water Level - Low Low, Level ~~2 and~~ Low Low ~~Low,~~ Level 1 provide sufficient protection for level transients in all other MODES. 2 6

5. Main Steam Isolation Valve - Closure

MSIV closure results in loss of the main turbine and the condenser as a heat sink for the nuclear steam supply system and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a Main Steam Isolation Valve - Closure signal before the MSIVs are completely closed in anticipation of the complete loss of

2

INSERT 7

since the scram occurs in the beginning of the event due to the loss of offsite power. However, analyses have been performed that indicate that the difference between a scram initiated at the beginning of the event and a scram initiated by Reactor Vessel Water Level - Low is negligible.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued) 2 9 6 Flow Referenced

the normal heat sink and subsequent overpressurization transient. However, for the overpressurization protection analysis of Reference 8, 6 the Average Power Range Monitor Fixed Neutron Flux - High Function, High along with the S/RVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is not assumed in the overpressurization analysis. main steam line break accident Additionally, MSIV closure is assumed in the transients analyzed in 12 Reference 8 (e.g., low steam line ~~pressure,~~ manual closure of MSIVs, high steam line flow). 2 The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

MSIV closure signals are initiated from position switches located on each of the eight MSIVs. Each MSIV has two position switches; one inputs to RPS trip system A while the other inputs to RPS trip system B. Thus, each RPS trip system receives an input from eight Main Steam Isolation Valve - Closure channels, each consisting of one position switch. The logic for the Main Steam Isolation Valve - Closure Function is arranged such that either the inboard or outboard valve on three or more of the main steam lines must close in order for a scram to occur.

The Main Steam Isolation Valve - Closure Allowable Value is specified to ensure that a scram occurs prior to a significant reduction in steam flow, thereby reducing the severity of the subsequent pressure transient.

Sixteen channels of the Main Steam Isolation Valve - Closure Function, with eight channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude the scram from this Function on a valid signal. This Function is only required in MODE 1 6 since, with the MSIVs open and the heat generation rate high, a with reactor pressure < 600 psig pressurization transient can occur if the MSIVs close. In MODE 2, the 6 heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection. and MODE 2 with reactor pressure ≥ 600 psig 6

This Function is automatically bypassed when the reactor mode switch is in a position other than run and the reactor pressure is < 600 psig.

6. Drywell Pressure - High

High pressure in the drywell could indicate a break in the RCPB. A reactor scram is initiated to minimize the possibility of fuel damage and to reduce the amount of energy being added to the coolant and the drywell. The Drywell Pressure - High Function is a secondary scram signal to Reactor Vessel Water Level - Low, Level 3 for LOCA events inside the 6 drywell. However, no credit is taken for a scram initiated from this

All changes are "2" unless otherwise noted

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

U

Function for any of the DBAs analyzed in the FSAR. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

switches

High drywell pressure signals are initiated from four pressure ~~transmitters~~ that sense drywell pressure. The Allowable Value was selected to be as low as possible and indicative of a LOCA inside primary containment.

Four channels of Drywell Pressure - High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required in MODES 1 and 2 where considerable energy exists in the RCS, resulting in the limiting transients and accidents.

7a, 7b. Scram Discharge Volume Water Level – High 1

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The SDV receives the water displaced by the motion of the CRD pistons during a reactor scram. Should this volume fill to a point where there is insufficient volume to accept the displaced water, control rod insertion would be hindered. Therefore, a reactor scram is initiated while the remaining free volume is still sufficient to accommodate the water from a full core scram. The two types of Scram Discharge Volume Water Level - High Functions are an input to the RPS logic. No credit is taken for a scram initiated from these Functions for any of the design basis accidents or transients analyzed in the FSAR. However, they are retained to ensure the RPS remains OPERABLE.

U

SDV water level is measured by two diverse methods. The level in each of the two SDVs is measured by two float type level switches and two thermal probes for a total of eight level signals. The outputs of these devices are arranged so that there is a signal from a level switch and a thermal probe to each RPS logic channel. The level measurement instrumentation satisfies the recommendations of Reference 9.

13

The Allowable Value is chosen low enough to ensure that there is sufficient volume in the SDV to accommodate the water from a full scram.

The Allowable Value refers to the volume of water in the discharge volume receiver tank and does not include the volume in the lines to the level switches.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Four channels of each type of Scram Discharge Volume Water Level - High Function, with two channels of each type in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from these Functions on a valid signal. These Functions are required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn. At all other times, this Function may be bypassed.

8. Turbine Stop Valve - Closure

Closure of the TSVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of the transients that would result from the closure of these valves. The Turbine Stop Valve - Closure Function is the primary scram signal for the turbine trip event analyzed in Reference 8. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded. 14 2

One

and two independent contacts

Turbine Stop Valve - Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve - Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve - Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, to consider this Function OPERABLE, the turbine bypass valves must remain shut at THERMAL POWER ≥ 30% RTP.

contacts

> 45 6

switches 2

INSERT 8

The Turbine Stop Valve - Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

2

INSERT 8

The pressure switches are normally adjusted lower (30% RTP) to account for the turbine bypass valves being opened, such that 14% of the THERMAL POWER is being passed directly to the condenser.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Eight channels of Turbine Stop Valve - Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function even if one TSV should fail to close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ~~≥ 30~~ > 45 % RTP. This Function is not required when THERMAL POWER is ~~< 30~~ ≤ 45 % RTP since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor ~~Fixed~~ Flow Referenced Neutron Flux - High High Functions are adequate to maintain the necessary safety margins. (6)

9. Turbine Control Valve Fast Closure, ~~Trip~~ Acceleration Relay Oil Pressure - Low (6)

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, ~~Trip~~ Oil Pressure - Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 8 (15). For this event, the reactor scram reduces the amount of energy required to be absorbed and~~, along with the actions of the EOC-RPT System,~~ ensures that the MCPR SL is not exceeded. (2)

Turbine Control Valve Fast Closure, ~~Trip~~ Acceleration Relay Oil Pressure - Low signals are initiated by ~~the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure transmitter is associated with each control valve, and the signal from each transmitter is assigned to a separate RPS logic channel.~~ INSERT 9 (6) (2) This Function must be enabled at THERMAL POWER ~~≥ 30~~ > 45 % RTP (6). This is normally accomplished automatically by pressure ~~transmitters~~ switches (2) sensing turbine first stage pressure~~; therefore, to consider this Function OPERABLE, the turbine bypass valves must remain shut at THERMAL POWER ≥ 30% RTP~~. INSERT 10 (2)

The Turbine Control Valve Fast Closure, ~~Trip~~ Acceleration Relay Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure. (6)

Four channels of Turbine Control Valve Fast Closure, ~~Trip~~ Acceleration Relay Oil Pressure - Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis

2

INSERT 9

loss of oil pressure at the acceleration relay. Two pressure switches are mounted on one pressure tap while two other pressure switches are mounted at a distance on another pressure tap. The pressure switches associated with one pressure tap are assigned to different RPS trip systems.

2

INSERT 10

The pressure switches are normally adjusted lower (30% RTP) to account for the turbine bypass valves being opened, such that 14% of the THERMAL POWER is being passed directly to the condenser.

All changes are "2" unless otherwise noted

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

> 45 ≤ 45

assumptions, whenever THERMAL POWER is ~~≥ 30~~% RTP. This Function is not required when THERMAL POWER is ~~< 30~~% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor ~~Fixed~~ Neutron Flux - High Functions are adequate to maintain the necessary safety margins. 6 6 6

Flow Referenced High

10. Reactor Mode Switch - Shutdown Position (A3 and B3)

The Reactor Mode Switch - Shutdown Position Function provides signals, via the two manual scram logic channels, ~~to each of the four RPS logic channels,~~ which are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

two

The reactor mode switch is a single switch with ~~four~~ channels, each of which provides input into one of the ~~RPS~~ logic channels.

two manual scram

There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on reactor mode switch position.

Two ~~Four~~ channels of Reactor Mode Switch - Shutdown Position Function, with one ~~two~~ channel~~s~~ in each trip system, are available and required to be OPERABLE. The Reactor Mode Switch - Shutdown Position Function is required to be OPERABLE in MODES 1 and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn. 6

11. Manual Scram (A3 and B3)

two

The Manual Scram push button channels provide signals, via the two manual scram logic channels, ~~to each of the four RPS logic channels,~~ which are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

two manual scram

There is one Manual Scram push button channel for each of the ~~four RPS~~ logic channels. In order to cause a scram it is necessary that ~~at least one~~ both channel s ~~in each trip system~~ be actuated.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Two ~~Four~~ channels of Manual Scram with one ~~two~~ channel~~s~~ in each trip system ~~arranged in a one-out-of-two logic~~ are available and required to be OPERABLE in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn. 6 10

ACTIONS

~~-----REVIEWER'S NOTE-----~~
~~Certain LCO Completion Times are based on approved topical reports. In order for a licensee to use the times, the licensee must justify the Completion Times as required by the staff Safety Evaluation Report (SER) for the topical report.~~ 7

A Note has been provided to modify the ACTIONS related to RPS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RPS instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RPS instrumentation channel.

A.1 and A.2

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (Ref. 16 ~~10~~) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function's inoperable channel is in one trip system and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2, and C.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the associated trip 6

BASES

ACTIONS (continued)

system must be placed in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternatively, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a full scram), Condition D must be entered and its Required Action taken. INSERT 11 2

B.1 and B.2

Condition B exists when, for any one or more Functions, at least one required channel is inoperable in each trip system. In this condition, provided at least one channel per trip system is OPERABLE, the RPS still maintains trip capability for that Function, but cannot accommodate a single failure in either trip system.

Required Actions B.1 and B.2 limit the time the RPS either scram logic, for any Function, would not accommodate single failure in ~~both~~ trip system~~s~~ (e.g., one-out-of-one and one-out-of-one arrangement for a typical four channel Function). The reduced reliability of this logic arrangement was not evaluated in Reference ~~10~~ 16 for the 12 hour Completion Time. Within the 6 hour allowance, the associated Function will have all required channels OPERABLE or in trip (or any combination) in one trip system. 1 2

Completing one of these Required Actions restores RPS to a reliability level equivalent to that evaluated in Reference ~~10~~ 16, which justified a 12 hour allowable out of service time as presented in Condition A. The trip system in the more degraded state should be placed in trip or, alternatively, all the inoperable channels in that trip system should be placed in trip (e.g., a trip system with two inoperable channels could be in a more degraded state than a trip system with four inoperable channels if the two inoperable channels are in the same Function while the four inoperable channels are all in different Functions). The decision of which trip system is in the more degraded state should be based on prudent judgment and take into account current plant conditions (i.e., what MODE the plant is in). If this action would result in a scram ~~or RPT~~, it is permissible to place the other trip system or its inoperable channels in trip. 2 2

2

INSERT 11

The 12 hour allowance is not allowed for Reactor Mode Switch - Shutdown Position Function and Manual Scram Function channels since with one channel inoperable RPS trip capability is not maintained. In this case, Condition C must be entered and its Required Actions taken.

BASES

ACTIONS (continued)

The 6 hour Completion Time is judged acceptable based on the remaining capability to trip, the diversity of the sensors available to provide the trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram.

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram [or ~~RPT~~]), Condition D must be entered and its Required Action taken. 3

INSERT 12 2

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if ~~multiple~~ one or more inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal. For the typical Function with one-out-of-two taken twice logic and the IRM and APRM Functions, this would require both trip systems to have one channel OPERABLE or in trip (or the associated trip system in trip). For Function 5 (Main Steam Isolation Valve - Closure), this would require both trip systems to have each channel associated with the MSIVs in three main steam lines (not necessarily the same main steam lines for both trip systems) OPERABLE or in trip (or the associated trip system in trip). 6

[For Function 8 (Turbine Stop Valve - Closure), this would require both trip systems to have three channels, each OPERABLE or in trip (or the associated trip system in trip). 1

INSERT 12A 6

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

2 **INSERT 12**

The 6 hour allowance is not allowed for Reactor Mode Switch - Shutdown Position Function and Manual Scram Function channels since with two channels inoperable RPS trip capability is not maintained. In this case, Condition C must be entered and its Required Action taken.

6 **INSERT 12A**

For Function 10 (Reactor Mode Switch - Shutdown Position) and Function 11 (Manual Scram), since each trip system only has one channel for each Function, with a channel inoperable, RPS trip capability is not maintained.

BASES

ACTIONS (continued)

D.1

Required Action D.1 directs entry into the appropriate Condition referenced in Table 3.3.1.1-1. The applicable Condition specified in the Table is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A, B, or C and the associated Completion Time has expired, Condition D will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1, F.1, and G.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. The allowed Completion Times are reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems. In addition, the Completion Time of Required Action E.1 is consistent with the Completion Time provided in LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)."

H.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by immediately initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are, therefore, not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

SURVEILLANCE REQUIREMENTS

~~-----REVIEWER'S NOTE-----~~
~~Certain Frequencies are based on approved topical reports. In order for a licensee to use these Frequencies, the licensee must justify the Frequencies as required by the staff SER for the topical report.~~ 7

BASES

SURVEILLANCE REQUIREMENTS (continued)

As noted at the beginning of the SRs, the SRs for each RPS instrumentation Function are located in the SRs column of Table 3.3.1.1-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains RPS trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. ~~4~~ 16) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary. 2

SR 3.3.1.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

~~The agreement criteria includes an expectation of one decade of overlap when transitioning between neutron flux instrumentation. The overlap between SRMs and IRMs must be demonstrated prior to withdrawing SRMs from the fully inserted position since indication is being transitioned from the SRMs to the IRMs. This will ensure that reactor power will not be increased into a neutron flux region without adequate indication. The overlap between IRMs and APRMs is of concern when reducing power into the IRM range (entry into MODE 2 from MODE 1). On power~~ 9

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~increases, the system design will prevent further increases (by initiating a rod block) if adequate overlap is not maintained. Overlap between IRMs and APRMs exists when sufficient IRMs and APRMs concurrently have onscale readings such that the transition between MODE 1 and MODE 2 can be made without either APRM downscale rod block or IRM upscale rod block. Overlap between SRMs and IRMs similarly exists when, prior to withdrawing the SRMs from the fully inserted position, IRMs are above mid-scale on Range 1 before SRMs have reached the upscale rod block.~~ 9

~~If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channels(s) declared inoperable. Only those appropriate channels that are required in the current MODE or condition should be declared inoperable.~~

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. ~~LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoints," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate within 2% RTP of calculated MFLPD.~~ The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.6. 6

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 25% RTP, the

All changes are "6" unless otherwise noted

SURVEILLANCE REQUIREMENTS (continued)

Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

~~SR 3.3.1.1.3~~

~~The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function uses the recirculation loop drive flows to vary the trip setpoint. This SR ensures that the total loop drive flow signals from the flow units used to vary the setpoint is appropriately compared to a calibrated flow signal and, therefore, the APRM Function accurately reflects the required setpoint as a function of flow. Each flow signal from the respective flow unit must be ≤ 105% of the calibrated flow signal. If the flow unit signal is not within the limit, one required APRM that receives an input from the inoperable flow unit must be declared inoperable.~~

~~The Frequency of 7 days is based on engineering judgment, operating experience, and the reliability of this instrumentation.~~

SR 3.3.1.1.~~4~~ 3

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specification tests at least once per refueling interval with applicable extensions. 1

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

BASES

SURVEILLANCE REQUIREMENTS (continued)

3

As noted, SR 3.3.1.1.~~4~~ is not required to be performed when entering MODE 2 from MODE 1, since testing of the MODE 2 required IRM ~~and APRM~~ Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

A Frequency of 7 days provides an acceptable level of system average unavailability over the Frequency interval and is based on reliability analysis (Ref. ~~9~~). 2

16

INSERT 13

SR 3.3.1.1.5

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification and non-Technical Specification tests at least once per refueling interval with applicable extensions. ~~In accordance with Reference 10, the scram contacts must be tested as part of the Manual Scram Function. A Frequency of 7 days provides an acceptable level of system average availability over the Frequency and is based on the reliability analysis of Reference 11. (The Manual Scram Function's CHANNEL FUNCTIONAL TEST Frequency was credited in the analysis to extend many automatic scram Functions' Frequencies.)~~ 1

2 INSERT 14

SR 3.3.1.1.6

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The ~~1000 MWD/T~~ 2000 effective full power hour Frequency is based on operating experience with LPRM sensitivity changes.

6 **INSERT 13**

SR 3.3.1.1.4

A functional test of each automatic scram contactor is performed to ensure that each automatic RPS logic channel will perform the intended function. There are four RPS channel test switches, one associated with each of the four automatic trip channels (A1, A2, B1, and B2). These test switches allow the operator to test the OPERABILITY of the individual trip logic channel automatic scram contactors as an alternative to using an automatic scram function trip. This is accomplished by placing the RPS channel test switch in the test position, which will input a trip signal into the associated RPS logic channel. The RPS channel test switches are not credited in the accident analysis, they just provide a method to test the automatic scram contactors. The Manual Scram Functions are not configured the same as the generic model used in Reference 16. However, Reference 16 concluded that the Surveillance Frequency extensions for RPS Functions were not affected by the difference in configuration since each automatic RPS logic channel has a test switch that is functionally the same as the manual scram switches in the generic model. As such, a functional test of each RPS automatic scram contactor using either its associated test switch or by test of any of the associated automatic RPS Functions is required to be performed once every 7 days. The Frequency of 7 days is based on the reliability analysis of Reference 16.

2 **INSERT 14**

The 31 day Frequency is based on engineering judgment, operating experience, and reliability of this instrumentation.

All changes are "6" unless otherwise noted

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.7 and SR 3.3.1.1.10

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification and non-Technical Specification tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. INSERT 14A The 92 day Frequency of SR 3.3.1.1.7 is based on the reliability analysis of Reference ~~10~~ 16. 1 2 2

The ~~18~~ 24 month Frequency of SR 3.3.1.1.10 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the ~~18~~ 24 month Frequency. 5

SR 3.3.1.1.8

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.1.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference ~~10~~ 16. 2

2

INSERT 14A

The CHANNEL FUNCTIONAL TEST (SR 3.3.1.1.10) for the Reactor Mode Switch - Shutdown Position channels will be performed by placing the reactor mode switch in the shutdown position.

All changes are "6" unless otherwise noted

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.9 and SR 3.3.1.1.11

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Note to SR 3.3.1.1.9 and to SR 3.3.1.1.11

APRM

2000 effective full power hours

Note ~~1~~ state~~s~~ that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the ~~1000 MWD/T~~ LPRM calibration against the TIPs (SR 3.3.1.1.6). ~~A second Note is provided~~ INSERT 15 (6)

Note 2 to SR 3.3.1.1.11

~~that~~ requires the ~~APRM and~~ IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 ~~APRM and~~ IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The Frequency of SR 3.3.1.1.9 is based upon the assumption of a 92 ~~184~~ day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.11 is based upon the assumption of a~~n~~ 24 ~~18~~ month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

~~SR 3.3.1.1.12~~

~~The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function uses an electronic filter circuit to generate a signal proportional to the core THERMAL POWER from the APRM neutron flux signal. This filter circuit is representative of the fuel heat transfer dynamics that produce the relationship between the neutron flux and the core THERMAL POWER. The Surveillance filter time constant must be verified to be ≤ 7 seconds to ensure that the channel is accurately reflecting the desired parameter.~~

~~The Frequency of 18 months is based on engineering judgment considering the reliability of the components.~~

6

INSERT 15

Changes in IRM neutron detector sensitivity are compensated for by periodically evaluating the compensating voltage setting and making adjustments as necessary.

All changes are "6" unless otherwise noted

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1 ~~13~~ 12

, "Control Rod OPERABILITY"

, "Scram Discharge Volume Vent and Drain Valves"

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods (LCO 3.1.~~3~~), and SDV vent and drain valves (LCO 3.1.~~8~~), overlaps this Surveillance to provide complete testing of the assumed safety function.

24

The ~~18~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 ~~18~~ month Frequency.

SR 3.3.1.1 ~~14~~ 13

Acceleration Relay

This SR ensures that scrams initiated from the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, ~~Trip~~ Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is > 45 ~~≥ 30~~% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed during in-service calibration at THERMAL POWER ~~≥ 30~~% RTP, if performing the calibration using actual turbine first stage pressure, to ensure that the calibration ~~remains~~ is valid. INSERT 16 2

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at > 45 ~~≥ 30~~% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, ~~Trip~~ Acceleration Relay Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

The Frequency of 24 ~~18~~ months is based on engineering judgment and reliability of the components.

2

INSERT 16

The pressure switches are normally adjusted lower (30% RTP) to account for the turbine bypass valves being opened, such that 14% of the THERMAL POWER is being passed directly to the condenser.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.~~15~~ 14 6

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. RPS RESPONSE TIME may be verified by actual response time measurements in any series of sequential, overlapping, or total channel measurements.

~~[------------------------------------REVIEWER'S NOTE------------------------------------ The following Bases are applicable for plants adopting NEDO-32291-A and/or Supplement 1. --]~~ 7

~~However, the sensors for Functions 3 and 4 are allowed to be excluded from specific RPS RESPONSE TIME measurement if the conditions of Reference 12 are satisfied. If these conditions are satisfied, sensor response time may be allocated based on either assumed design sensor response time or the manufacturer's stated design response time. When the requirements of Reference 12 are not satisfied, sensor response time must be measured. Furthermore, measurement of the instrument loops response times for Functions 3 and 4 is not required if the conditions of Reference 13 are satisfied.]~~ 2

The RPS RESPONSE TIME acceptance ~~criteria are included in Reference 11~~ criterion is 50 milliseconds.

~~As noted, neutron detectors are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time.~~ 6

A RPS RESPONSE TIME tests are conducted on ~~an 18~~ 24 month ~~STAGGERED TEST BASIS~~. Note ~~2~~ requires STAGGERED TEST BASIS Frequency to be determined based on 4 channels per trip system, in lieu of the 8 channels specified in Table 3.3.1.1-1 for the MSIV - Closure Function. This Frequency is based on the logic interrelationships of the various channels required to produce an RPS scram signal. The ~~18~~ 24 month Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. 6 1 6

BASES

REFERENCES

1. Regulatory Guide 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation."

U Section 7.6.1.2.1

2. ~~F~~SAR, ~~Figure [ ]~~. (2)

INSERT 17

~~3. FSAR, Section [15.1.2].~~

USAR, Section 7.3.4.3.

7 ~~4~~. ~~NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.~~ (2)

~~5. FSAR, Section [5.2.2].~~

~~6. FSAR, Section [15.1.38].~~

~~7. FSAR, Section [6.3.3].~~

~~8. FSAR, Chapter [15].~~

INSERT 18 (2)

13 ~~9~~. P. Check (NRC) letter to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980. (2)

INSERT 19

16 ~~10~~. C NED~~O~~-30851-P-A , "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988. (2)

~~11. FSAR, Table [7.2-2].~~

~~[12. NEDO-32291-A, "System Analyses for the Elimination of Selected Response Time Testing Requirements," October 1995.~~ (7)

~~13. NEDO-32291-A, Supplement 1, "System Analyses for the Elimination of Selected Response Time Testing Requirements," October 1999.]~~

2 **INSERT 17**

3. USAR, Section 7.6.1.2.5.

4. USAR, Chapter 14.

5. USAR, Chapter 14A.

6. USAR, Section 7.8.2.1.

2 **INSERT 18**

8. USAR, Section 14.6.

9. USAR, Section 14.5.1.

10. USAR, Section 14.7.1.

11. USAR, Section 14.7.2.

12. USAR, Section 14.7.3.

2 **INSERT 19**

14. USAR, Section 14.4.5.

15. USAR, Section 14.4.1.

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Table B 3.3.1.1-1 (page 1 of 1)
RPS Instrumentation Sensor Diversity

Initiation Events	Scram Sensors for Initiating Events						
	RPV Variables			Anticipatory			Fuel
	(a)	(b)	(c)	(d)	(e)	(f)	(g)
MSIV Closure	x		x			x	x
Turbine Trip (w/bypass)	x			x	x		x
Generator Trip (w/bypass)	x			x			x
Pressure Regulator Failure (primary pressure decrease) (MSIV closure trip)	x	x	x			x	x
Pressure Regulator Failure (primary pressure decrease) (Level 8 trip)	x				x		x
Pressure Regulator Failure (primary pressure increase)	x						x
Feedwater Controller Failure (high reactor water level)	x	x			x		x
Feedwater Controller Failure (low reactor water level)	x		x			x	
Loss of Condenser Vacuum	x				x	x	x
Loss of AC Power (loss of transformer)	x		x		x	x	
Loss of AC Power (loss of grid connections)	x		x	x	x	x	x

(a) Reactor Vessel Steam Dome Pressure - High
(b) Reactor Vessel Water Level - High, Level 8
(c) Reactor Vessel Water Level - Low, Level 3
(d) Turbine Control Valve Fast Closure
(e) Turbine Stop Valve - Closure
(f) Main Steam Isolation Valve - Closure
(g) Average Power Range Monitor Neutron Flux - High

-----REVIEWER'S NOTE-----

This Table for illustration purposes only.

4

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.1.1 BASES, REACTOR PROTECTION SYSTEM (RPS) INSTRUMENTATION

1. Grammatical/editorial error corrected.

2. Changes are made (additions, deletions, and/or changes) to the ISTS Bases, which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. Table B 3.3.1.1-1 has been deleted since it provides generic, not plant specific types of information. The information in the Table could be misleading as to which plant specific analyses take credit for these channels to perform a function during accident and transient scenarios. All references to this Table have been deleted. This deletion is consistent with many other BWR ITS conversions (e.g., Quad Cities 1 and 2, LaSalle 1 and 2, Dresden 2 and 3).

5. Changes are made to reflect the Specification.

6. Changes are made to reflect changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

7. This Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

8. The Title's of the LCO's have been included the first time it appears in the LCO Bases to be consistent with other places in the Bases.

9. The Bases discussion associated with the SRM/IRM and IRM/APRM overlap has been deleted. This Bases requires an overlap check to be performed between the SRMs and IRMs during a startup (i.e., prior to withdrawing SRMs from the fully inserted position) and verifying a one decade overlap. The Bases also requires an overlap check to be performed between the APRMs and IRMs during a shutdown (i.e., during entry into MODE 2 from MODE 1) and verifying a one decade overlap. These requirements are not consistent with the CHANNEL CHECK requirement in the actual ITS Surveillance Requirement. The definition of CHANNEL CHECK states "A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter." The definition does not require an overlap check of any type of channel. This overlap check described in the Bases is an additional requirement, over and above the requirements in the actual Surveillance Requirement.

10. The logic description has been deleted since it is described in a previous paragraph.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.1.1, REACTOR PROTECTION SYSTEM (RPS) INSTRUMENTATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.14

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1433.

CTS Table 4.1.1 requires a weekly functional test of the Manual Scram Function. ITS Table 3.3.1.1-1 Function 11 and ITS SR 3.3.1.1.5 requires the performance of the same test at a 31 day Frequency. This changes the CTS by extending the Manual Scram functional test Frequency from 7 days to 31 days.

The purpose of the functional test is to ensure the Manual Scram Function instrumentation is functioning properly. This changes the CTS by extending the requirement to perform the test from 7 days to 31 days. The Manual Scram functional test Frequency was previously changed from monthly to weekly as part of the amendment request that adopted GE Topical Report NEDC-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," dated March 1988. NEDC-30851-P-A performed an analysis to extend the CHANNEL FUNCTIONAL TEST Frequency of the automatic RPS channels from monthly to quarterly. In order to justify this extension, it was necessary to actuate the automatic logic scram relays every 7 days. Therefore, NEDC-30851-P-A also changed the CHANNEL FUNCTIONAL TEST for the Manual Scram Function from monthly to weekly since, for the four manual scram pushbutton RPS design, performing a CHANNEL FUNCTIONAL TEST of the Manual Scram Functions (i.e., the manual pushbuttons) actuates the automatic logic scram relays. The Monticello amendment request adopting NEDC-30851-P-A included changing the Manual Scram functional test Frequency from monthly to weekly, and was approved by the NRC in License Amendment 81, dated April 16, 1992. However, the Manual Scram pushbuttons at Monticello do not actuate the automatic logic scram relays; a separate manual scram logic channel (designated A3 and B3) for each of the two manual scram pushbuttons is provided. Therefore, NEDC-30851-P-A did not actually require a change to the Manual Scram functional test frequency. To ensure the automatic logic scram relays are tested every week, the CTS Bases was updated on June 10, 2004 and clarifies that the Manual Scram refers to a manually initiated trip of both the Manual and Auto Scram logic. However, this change was made just to ensure the requirements of NEDC-30851-P-A, as they relate to the automatic logic scram relays, were met. ITS SR 3.3.1.1.4 has been included to ensure the automatic logic scram relays are tested every week. A review of past Manual Scram functional test Surveillances was performed and all completed tests were successful. Both monthly and weekly tests performed in 1992 (pre- and post-implementation of the monthly to weekly Surveillance Frequency change) and recent weekly tests were reviewed. In total, 27 completed Surveillances were reviewed and the Manual Scram functional test was successful in every case. Furthermore, the Manual Scram functional test only includes switches and relays and does not rely on instrument setpoints or other calibrations that are potentially subject to drift. Therefore, a monthly functional test Frequency for the Manual Scram and continued weekly testing of the automatic logic scram relays (to support NEDC-30851-P-A implementation) is acceptable. This change

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.1.1, REACTOR PROTECTION SYSTEM (RPS) INSTRUMENTATION

is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change extends the requirement to perform the Manual Scram functional test from 7 days to 31 days. The relaxed Surveillance Frequency has been established based on achieving acceptable levels of equipment reliability. Consequently, equipment which could initiate an accident previously evaluated will continue to operate as expected, and the probability of the initiation of any accident previously evaluated will not be significantly increased. The equipment being tested is still required to be OPERABLE and capable of performing any accident mitigation functions assumed in the accident analyses. As a result, the consequences of any accident previously evaluated are not increased significantly. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change extends the requirement to perform the Manual Scram functional test from 7 days to 31 days. This change will not physically alter the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change extends the requirement to perform the Manual Scram functional test from 7 days to 31 days. As provided in the discussion of change, the relaxation in the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Thus, appropriate equipment continues to be tested at a Frequency that gives confidence that the equipment can perform its assumed safety function when required. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.1.1, REACTOR PROTECTION SYSTEM (RPS) INSTRUMENTATION

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

ATTACHMENT 2

ITS 3.3.1.2, Source Range Monitor (SRM) Instrumentation

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS | ITS

Add proposed Table 3.3.1.2-1 footnote (b) L.1

Source Range Monitor (SRM) Instrumentation

3.0 LIMITING CONDITIONS FOR OPERATION | **4.0 SURVEILLANCE REQUIREMENTS**

3.3.1.2

B. Core Monitoring

Table 3.3.1.2-1, SR 3.3.1.2.2 Note 1 — During core alterations two SRM's shall be operable,

Table 3.3.1.2-1 — one in and one adjacent to any core quadrant where fuel

SR 3.3.1.2.2 — or control rods are being moved. For an SRM to be considered operable, the following conditions shall be satisfied:

Add proposed SR 3.3.1.2.2.a and SR 3.3.1.2.2 Note 2 (L.1)

Add proposed SR 3.3.1.2.2 Frequency (M.3)

1. ~~The SRM shall be inserted to the normal operating level.~~ (LA.1) (Use of special moveable, dunking type detectors during initial fuel loading and major core alterations is permissible as long as the detector is connected into the normal SRM circuit.) — Table 3.3.1.2-1 footnote (c)

2. The SRM shall have a minimum of 3 CPS with all rods fully inserted in the core except when both of the following conditions are fulfilled: — SR 3.3.1.2.4

   and signal to noise ratio ≥ 3:1 (A.2)

   a. No more than two fuel assemblies are present in the core quadrant associated with the SRM, — SR 3.3.1.2.4 Note

   b. While in core, these fuel assemblies are in locations adjacent to the SRM.

B. Core Monitoring (M.1) (M.2)

Table 3.3.1.2-1 — Prior to making any alterations to the core while more

SR 3.3.1.2.4 Note — than two fuel assemblies are present in any reactor quadrant, the SRM's shall be functionally tested and

SR 3.3.1.2.5 — checked for neutron response. Thereafter, the SRM's

SR 3.3.1.2.4 — will be checked daily for response.

and determine signal to noise ratio (M.1)

every 7 days (M.1)

every 12 hours (M.2)

Add proposed SR 3.3.1.2.1 and SR 3.3.1.2.7 for MODE 5 (M.4)

C. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool, the pool water level shall be maintained at a level of greater or equal to 33 feet.

C. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool the pool level shall be recorded daily.

See ITS 3.7.8

D. The reactor shall be shutdown for a minimum of 24 hours prior to movement of fuel within the reactor.

See CTS 3/4.10.D, in ITS Section 3.9

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A.1

ITS

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
(b) Whenever the reactor is in the startup or run mode below 10% rated thermal power, no control rods shall be moved unless the rod worth minimizer is operable or a second independent operator or engineer verifies that the operator at the reactor console is following the control rod program. The second operator may be used as a substitute for an inoperable rod worth minimizer during a startup only if the rod worth minimizer fails after withdrawal of at least twelve control rods.	(iv) The rod block function of the rod worth minimizer shall be verified by attempting to withdraw an out-of-sequence control rod beyond the block point. (b) If the rod worth minimizer is inoperable while the reactor is in the startup or run mode below 10% rated thermal power and the second independent operator or engineer is being used, he shall verify that all rod positions are correct prior to commencing withdrawal or insertion of each rod group.

See ITS 3.3.2.1

M.7 Add proposed Table 3.3.1.2-1 footnote (a)

Table 3.3.1.2-1 — 4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels
LCO 3.3.1.2 Table 3.3.1.2-1 — have an observed count rate equal to or greater than three counts per second.

three in MODE 2 — M.7

Applicability — 4. Prior to control rod withdrawal for startup or during refueling verify that at least two source range channels
SR 3.3.1.2.4 — have an observed count rate of at least three counts per second.

Add proposed 12 hour and 24 hour Frequency — M.2

Add proposed MODES 3 and 4 SRM requirements — M.6

Add proposed SR 3.3.1.2.3, SR 3.3.1.2.4, SR 3.3.1.2.6, and SR 3.3.1.2.7 for MODES 3 and 4 — M.6

Add proposed SR 3.3.1.2.1, SR 3.3.1.2.6, and SR 3.3.1.2.7 for MODE 2 — M.4

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A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
F. Scram Discharge Volume	F. Scram Discharge Volume

1. During reactor operation, the scram discharge volume vent and drain valves shall be operable, except as specified below.

2. If any scam discharge volume vent or drain valve is made or found inoperable, the integrity of the scram discharge volume shall be maintained by either:

   a. Verifying daily, for a period not to exceed 7 days, the operability of the redundant valve(s), or

   b. Maintaining the inoperable valve(s), or the associated redundant valve(s), in the closed position. Periodically the inoperable and the redundant valve(s) may both be in the open position to allow draining the scram discharge volume.

   If a or b above cannot be met, at least all but one operable control rods (not including rods removed per specification 3.10.E or inoperable rods allowed by 3.3.A.2) shall be fully inserted within ten hours.

The scram discharge volume vent and drain valves shall be cycled quarterly.

Once per operating cycle verify the scram discharge volume vent and drain valves close within 30 seconds after receipt of a reactor scram signal and open when the scram is reset.

{See ITS 3.1.8}

{See ITS 3.1.1}

(except when the reactor mode switch is in the Refuel position)

A.3

G. Required Action

A.3

ACTION C

1. *If Specifications 3.3.A through D above are not met, an orderly shutdown shall be initiated and have reactor in the cold shutdown condition within 24 hours.*

Add proposed ACTIONS A and B

L.2

Add proposed ACTION E for MODE 5

M.5

MODE 3 in 12 hours

M.8

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DISCUSSION OF CHANGES
ITS 3.3.1.2, SOURCE RANGE MONITOR (SRM) INSTRUMENTATION

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.10.B.2 states that an OPERABLE SRM shall have a minimum of 3 CPS. ITS SR 3.3.1.2.4 requires verification that the SRM count rate is ≥ 3 CPS and also requires that the signal to noise ratio is ≥ 3:1. This changes the CTS by adding a requirement to verify the SRM signal to noise ratio is within limit.

The purpose of CTS 3.10.B is to specify the minimum count rate required to verify the associated SRM channel is OPERABLE. Similarly, ITS SR 3.3.1.2.4 requires verification of the SRM count rate but also requires verification that the SRM signal to noise ratio is ≥ 3 CPS. This change is acceptable because the current requirement for the SRM to be ≥ 3 CPS is based upon a signal to noise ratio ≥ 3:1. This change is administrative because it does not result in a technical change to the CTS.

A.3 These changes to CTS 3.3.G are provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the NRC for approval in NMC letter L-MT-05-013, from Thomas J. Palmisano (NMC) to USNRC, dated April 12, 2005. As such, these changes are administrative.

MORE RESTRICTIVE CHANGES

M.1 CTS 4.10.B, in part, requires performance of a functional test prior to making any alterations to the core. ITS Table 3.3.1.2-1 requires a CHANNEL FUNCTIONAL TEST (ITS SR 3.3.1.2.5) every 7 days when in MODE 5. Additionally, ITS SR 3.3.1.2.5 requires determination of the signal to noise ratio (unless there are less than or equal to two fuel assemblies adjacent to the SRM and no other fuel assemblies are in the associated core quadrant). This changes the CTS by requiring the CHANNEL FUNCTIONAL TEST every 7 days when in MODE 5, not just prior to the start of CORE ALTERATIONS, and by requiring an additional Surveillance requirement to verify the signal to noise ratio (unless there are less than or equal to two fuel assemblies adjacent to the SRM and no other fuel assemblies are in the associated core quadrant) every 7 days.

The purpose of CTS 4.10.B, in part, is to demonstrate the associated SRM channel will function properly during CORE ALTERATIONS. Similarly, ITS SR 3.3.1.2.5 requires performance of a CHANNEL FUNCTIONAL TEST. In addition, ITS SR 3.3.1.2.5 requires a determination of the signal to noise ratio (unless there are less than or equal to two fuel assemblies adjacent to the SRM and no other fuel assemblies are in the associated core quadrant) and specifies

DISCUSSION OF CHANGES
ITS 3.3.1.2, SOURCE RANGE MONITOR (SRM) INSTRUMENTATION

a Frequency of 7 days. The Applicability requirements of ITS Table 3.3.1.2-1 require ITS SR 3.3.1.2.5 in MODE 5, the MODE in which CORE ALTERATIONS would be performed. This change is acceptable because it retains the requirement to perform a CHANNEL FUNCTIONAL TEST consistent with the conditions required by CTS 4.10.B, verifies the integrity of the signal by determining the signal to noise ratio, and ensures continued proper function by imposing a Frequency to periodically perform the CHANNEL FUNCTIONAL TEST. This change is designated as more restrictive because the ITS adds a Surveillance Requirement and a periodic Frequency to the CHANNEL FUNCTIONAL TEST that is not currently required by the CTS.

M.2 CTS 4.10.B, in part, requires a check of SRM neutron response prior to making any alterations to the core and daily thereafter. CTS 4.3.B.4 requires, prior to control rod withdrawal for startup or during refueling, that the SRM count rate be ≥ 3 CPS. ITS Table 3.3.1.2-1 requires, when in MODE 5, a verification of the count rate (ITS SR 3.3.1.2.4) every 12 hours during CORE ALTERATIONS and every 24 hours at all other times. ITS Table 3.3.1.2-1 also requires, when in MODE 2 with IRMs on Range 2 or below, a verification of count rate (ITS SR 3.3.1.2.4) every 24 hours. This changes the CTS by requiring an increased Surveillance Frequency during CORE ALTERATIONS requiring a count rate verification anytime when in MODE 5, not just during CORE ALTERATIONS, and a count rate verification in MODE 2, not just prior to entering MODE 2.

The purpose of CTS 4.10.B, in part, and CTS 3.3.B.4, is to verify the associated SRM channel is functioning properly during CORE ALTERATIONS and prior to a reactor startup. Similarly, ITS SR 3.3.1.2.4 requires verification of the count rate every 24 hours when CORE ALTERATIONS are not in progress but increases the Frequency to 12 hours during CORE ALTERATIONS. The Applicability requirements of ITS Table 3.3.1.2-1 require ITS SR 3.3.1.2.4 in MODE 5, the MODE in which CORE ALTERATIONS would be performed, and MODE 2 with IRMs in Range 2 or below, the MODE in which reactor startup occurs. This change is acceptable because it retains the requirement to perform a check of the neutron response consistent with the modes and conditions required by CTS 4.10.B and CTS 4.3.B.4 and verifies continued proper neutron response at an increased Frequency during CORE ALTERATIONS. This change is more restrictive because the ITS requires the Surveillance to be performed more frequently than is currently required by the CTS.

M.3 CTS 3.10.B specifies location requirements for SRMs during CORE ALTERATIONS. ITS SR 3.3.1.2.2 requires verification of SRM locations and specifies a Frequency every 12 hours. This changes the CTS by providing a specific Surveillance Frequency.

The purpose of CTS 3.10.B is to ensure an OPERABLE SRM is located in the core regions where CORE ALTERATIONS are taking place. Similarly, ITS SR 3.3.1.2.2 requires verification that an OPERABLE SRM is located in the fuel region where CORE ALTERATIONS are being performed and also specifies a Frequency of 12 hours during CORE ALTERATIONS. This change is acceptable because it retains the requirement to verify an OPERABLE SRM detector is located in the required core region and requires verification at a specified Frequency during CORE ALTERATIONS. This change is more restrictive

DISCUSSION OF CHANGES
ITS 3.3.1.2, SOURCE RANGE MONITOR (SRM) INSTRUMENTATION

because the ITS specifies a Surveillance Frequency not currently required by the CTS.

M.4 The CTS does not require a CHANNEL CHECK or a CHANNEL CALIBRATION of the SRMs while in MODE 2 or 5, and does not require a CHANNEL FUNCTIONAL TEST of the SRMs while in MODE 2. ITS SR 3.3.1.2.1 requires performance of a CHANNEL CHECK every 12 hours in MODES 2 and 5. ITS SR 3.3.1.2.6 requires performance of a CHANNEL FUNCTIONAL TEST including a signal to noise ratio determination every 31 days while in MODE 2. ITS SR 3.3.1.2.7 requires performance of a CHANNEL CALIBRATION every 24 months in MODES 2 and 5. This changes the CTS by adding new Surveillance Requirements.

The purpose of ITS SR 3.3.1.2.1 is to ensure that gross failure of the SRM has not occurred. The purpose of ITS SR 3.3.1.2.6 is to verify performance of the SRM channel and is modified by a Note. The Note allows the Surveillance to be delayed until entry into the specified condition of the Applicability and allows 12 hours after entering MODE 2 with IRMs on Range 2 or below. This allowance is based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance at higher power levels. The purpose of ITS SR 3.3.1.2.7 is to verify performance of the SRM and associated circuitry and is modified by two Notes. Note 1 excludes the neutron detectors for the CHANNEL CALIBRATION because they cannot be readily adjusted. Note 2 allows the Surveillance to be delayed until entry into the specified condition of the Applicability and allows 12 hours after entering MODE 2 with IRMs on Range 2 or below. This allowance is based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance at higher power levels. The addition of the CHANNEL CHECK is acceptable because the Surveillance will detect gross channel failure and is key to verifying the instrumentation operates properly between CHANNEL CALIBRATIONS. The addition of a Surveillance to perform a CHANNEL FUNCTIONAL TEST every 31 days will ensure proper operation of the SRM channels. The addition of a Surveillance to perform a CHANNEL CALIBRATION every 24 months will ensure proper operation of the SRM circuitry. This change is more restrictive because the ITS requires Surveillance Requirements not currently required by the CTS.

M.5 CTS 3.10.B does not specify any Actions for an inoperable required SRM during CORE ALTERATIONS. CTS 3.3.G.1 specifies that if CTS 3.3.B.4 requirements are not met, the reactor shall be placed in the cold shutdown condition within 24 hours. Thus, when a required SRM is inoperable during control rod withdrawal in refuel (i.e., CORE ALTERATIONS) the CTS 3.3.G.1 requirement to be in cold shutdown in 24 hours would apply. However, since the unit is already in refuel the action to be in cold shutdown is not required (i.e., no ACTIONS are actually applicable). When one or more SRMs are inoperable in MODE 5, ITS 3.3.1.2 ACTION E requires CORE ALTERATIONS, except for control rod insertion, be immediately suspended, and action be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. This changes the CTS by specifying actions that are necessary to prevent reactivity changes and ensure the reactor will be at its minimum reactivity.

DISCUSSION OF CHANGES
ITS 3.3.1.2, SOURCE RANGE MONITOR (SRM) INSTRUMENTATION

ITS 3.3.1.2 ACTION E provides actions to immediately suspend CORE ALTERATIONS (except for control rod insertion), and immediately initiate action to insert all insertable control rods in cells containing one or more fuel assemblies. This change is acceptable because suspending CORE ALTERATIONS prevents the two most probable causes of reactivity changes, fuel loading and control rod withdrawal, from occurring, and inserting all insertable control rods ensures that the reactor will be at its minimum reactivity given that fuel is present in the core. This change is more restrictive because the ITS adds Required Actions not currently required by the CTS.

M.6 The CTS does not provide any SRM requirements in MODES 3 and 4 (i.e., Hot and Cold Shutdown). ITS Table 3.3.1.2-1 includes requirements for two SRMs to be OPERABLE in MODES 3 and 4 and specifies the applicable Surveillances required to demonstrate the SRMs OPERABILITY. The Surveillance Requirements in MODES 3 and 4 are ITS SR 3.3.1.2.3 (CHANNEL CHECK), ITS SR 3.3.1.2.4 (count rate verification), ITS SR 3.3.1.2.6 (CHANNEL FUNCTIONAL TEST and signal to noise ratio determination), and ITS SR 3.3.1.2.7 (CHANNEL CALIBRATION). In addition, ITS 3.3.1.2 ACTION D is added to address one or more inoperable SRMs in MODE 3 or 4. This changes the CTS by requiring two SRMs to be OPERABLE in MODES 3 and 4, and adding the Surveillances and ACTIONS associated with the added Applicability.

The purpose of the ITS Table 3.3.1.2-1 requirements for two SRMs to be OPERABLE in MODES 3 and 4 is to provide neutron flux indicated to the operator during shutdown conditions. This change is acceptable because the requirement for SRMs to be OPERABLE in MODES 3 and 4 provide the primary indication of neutron flux level in these MODES. The addition of Surveillances verify the SRMs are OPERABLE and the ACTION ensures the reactor will be at its minimum reactivity level and that control rod withdrawal is prevented. This change is more restrictive because the ITS LCO will be applicable under more reactor operating conditions than required in the CTS.

M.7 CTS 3.3.B.4 and CTS 4.3.B.4 states, in part, that two SRMs are required to be OPERABLE when control rods are being withdrawn for startup. ITS Table 3.3.1.2-1 requires three SRMs to be OPERABLE in MODE 2 (Startup). Furthermore, the SRMs are required to be OPERABLE only when IRMs are on Range 2 or below. This changes the CTS by requiring 3 SRMs to be OPERABLE in MODE 2 when IRMs are on Range 2 or below.

The purpose of CTS 3.3.B.4 and CTS 4.3.B.4 is to ensure sufficient SRMs are available to provide neutron flux indication to the operator during startup. Similarly, ITS Table 3.3.1.2-1 requires three SRMs to be OPERABLE in MODE 2 with IRMs on Range 2 or below (Footnote (a)). This change is acceptable because the requirement for three SRMs to be OPERABLE in MODE 2 provides a representation of the overall core response during those periods when reactivity changes are occurring throughout the core. This change is more restrictive because the ITS will require more SRMs to be OPERABLE in MODE 2 than is currently required in the CTS. Furthermore, the ITS does not specify how long after the control rods are withdrawn that the SRMs must remain OPERABLE. Thus the addition of Footnote (a) is also more restrictive.

DISCUSSION OF CHANGES
ITS 3.3.1.2, SOURCE RANGE MONITOR (SRM) INSTRUMENTATION

M.8 CTS 3.3.G.1, in part, requires the unit to be in cold shutdown in 24 hours if the conditions of CTS 3.3.B.4 are not met. ITS 3.3.1.2 ACTION C requires the unit to be in MODE 3 within 12 hours if the Required Action and Completion Time of Condition A or B are not met. This changes the CTS by requiring the plant to be in MODE 3 in 12 hours in lieu of being in MODE 4 in 24 hours.

The purpose of CTS 3.3.G.1 is to place the unit in a condition outside the Applicability of the Specification. While CTS 3.3.G.1 requires a shutdown to MODE 4, in actuality, only a shutdown to a condition where no positive reactivity changes are possible is required (i.e., MODE 3). ITS 3.3.1.2 ACTION C requires the unit to be in MODE 3 within 12 hours if any Required Action and Completion Time of Condition A or B (see DOC L.2) are not met. In MODE 3 with the reactor mode switch in shutdown, subsequent control rod withdrawal is prevented by maintaining a control rod block. The Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 in an orderly manner and without challenging unit systems. This change is acceptable because it requires the unit to be in an intermediate condition (MODE 3) sooner than is currently required (12 hours versus 24 hours). This portion of the change reduces the time the unit would be allowed to continue to operate in MODE 2 once the condition is identified. This change is more restrictive because less time is allowed to shut down the plant in the ITS than in the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 2 – Removing Details of System Design and System Description, including Design Limits)* CTS 3.10.B.1 requires SRMs be inserted to normal operating level. ITS 3.3.2.1 does not specify the level to which SRMs are required to be inserted. This changes the CTS by moving the information of the SRM insertion level to the ITS Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the SRMs to be OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is a less restrictive removal of detail because information relating to system design is being removed from the Technical Specifications.

DISCUSSION OF CHANGES
ITS 3.3.1.2, SOURCE RANGE MONITOR (SRM) INSTRUMENTATION

LESS RESTRICTIVE CHANGES

L.1 *(Category 6 – Relaxation Of Surveillance Requirement Acceptance Criteria)* CTS 3.10.B requires two SRMs to be OPERABLE during core alterations, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant. ITS SR 3.3.1.2.2.a ensures an OPERABLE SRM is in the fueled region, and Note 2 to ITS SR 3.3.1.2.2 adds an allowance that one SRM may be used to satisfy more than one SRM location requirement. ITS Table 3.3.1.2-1 Footnote (b) allows the number of SRM channels required to be OPERABLE to be reduced from two to one "during spiral offload or reload when the fueled region includes only that SRM detector." This changes the CTS by requiring only one SRM to be OPERABLE during CORE ALTERATIONS that encompass special offloading and reloading when the fueled region includes only that SRM detector.

The purpose of CTS 3.10.B is to require sufficient SRMs to be OPERABLE to ensure adequate neutron flux monitoring in the fueled regions of the core. Similarly, ITS SR 3.3.1.2.2.a requires verification that an OPERABLE SRM is in the fueled region. ITS SR 3.3.1.2.2 Note 2 provides an allowance that one SRM may be used to satisfy more than one SRM location requirement. In addition, if a spiral offload or reload pattern is used, ITS Table 3.3.1.2-1 Footnote (b) allows a reduction in the number of required OPERABLE SRM channels. This change is acceptable because the requirement to ensure adequate flux monitoring in the fueled regions of the core is retained, clarification of the use of the one SRM to satisfy more than one location requirement is provided, and allowance for a reduction from two to one OPERABLE SRMs is provided consistent with the requirement to maintain an OPERABLE SRM in the fueled region since the use of a spiral pattern provides assurance that the OPERABLE SRM is in the optimum position for monitoring changes in neutron flux levels resulting from the CORE ALTERATION. This change is less restrictive because the ITS allows a reduction in required OPERABLE SRMs not currently allowed in the CTS.

L.2 *(Category 4 – Relaxation of Required Action)* CTS 3.3.G.1, in part, requires the unit to be in cold shutdown in 24 hours if the conditions of CTS 3.3.B.4 are not met. ITS 3.3.1.2 ACTIONS A and B provide allowances to restore inoperable SRMs in MODE 2 with the IRMs on Range 2 or below prior to requiring a unit shutdown. ITS 3.3.1.2 ACTION A allows 4 hours to restore one or more inoperable required SRM channels to OPERABLE. Furthermore, ITS 3.3.1.2 ACTION B requires immediate suspension of all control rod withdrawal if there are no OPERABLE required SRMs. This changes the CTS by providing an allowance to restore inoperable SRMs, in MODE 2 with IRMs on Range 2 or below, before requiring a unit shutdown.

The purpose of CTS 3.3.G.1 is to place the unit in a condition outside the Applicability of the Specification. In MODE 2 with IRMs on Range 2 or below, SRMs provide the only means of monitoring core reactivity and criticality. With any number of the required SRMs inoperable, the ability to monitor neutron flux is degraded. Therefore, a limited time is allowed to restore the inoperable channels to OPERABLE status. ITS 3.3.1.2 ACTION A allows control rod withdrawal to continue for up to 4 hours with less than the required number of SRMs OPERABLE; and may be exited either by restoration of the required number of

DISCUSSION OF CHANGES
ITS 3.3.1.2, SOURCE RANGE MONITOR (SRM) INSTRUMENTATION

SRM channels or by increasing reactor power until the IRMs are above Range 2. ITS 3.3.1.2 ACTION B requires suspending control rod withdrawal immediately, thus preventing any positive changes in reactivity. These changes are acceptable because: a) SRMs are not credited in the analysis of any accident and exist solely to allow operators to monitor changes in power level during startup; b) at least one SRM will remain OPERABLE during any rod withdrawal; c) excessive reactivity additions during MODE 2 will be quickly identified and mitigated by the IRMs and; d) the analysis assumptions are not affected by the operator's ability to monitor changes in flux levels. This change is less restrictive because less stringent Required Actions are being applied in ITS than were applied in CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.3 INSTRUMENTATION

3.3.B.4, 3.10.B — 3.3.1.2 Source Range Monitor (SRM) Instrumentation

M.6 — LCO 3.3.1.2 The SRM instrumentation in Table 3.3.1.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.2-1.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
3.3.G.1	A. One or more required SRMs inoperable in MODE 2 with intermediate range monitors (IRMs) on Range 2 or below.	A.1 Restore required SRMs to OPERABLE status.	4 hours	
3.3.G.1	B. [Three] required SRMs inoperable in MODE 2 with IRMs on Range 2 or below.	B.1 Suspend control rod withdrawal.	Immediately	1
3.3.G.1	C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	12 hours	
DOC M.6	D. One or more required SRMs inoperable in MODE 3 or 4.	D.1 Fully insert all insertable control rods. AND D.2 Place reactor mode switch in the shutdown position.	1 hour 1 hour	

CTS

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M.5	E. One or more required SRMs inoperable in MODE 5.	E.1 Suspend CORE ALTERATIONS except for control rod insertion.	Immediately
		AND	
		E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

---NOTE---
Refer to Table 3.3.1.2-1 to determine which SRs apply for each applicable MODE or other specified conditions.

	SURVEILLANCE		FREQUENCY
DOC M.4	SR 3.3.1.2.1	Perform CHANNEL CHECK.	12 hours

CTS

SURVEILLANCE REQUIREMENTS (continued)

CTS		SURVEILLANCE	FREQUENCY	
3.10.B	SR 3.3.1.2.2	---NOTES--- 1. Only required to be met during CORE ALTERATIONS. 2. One SRM may be used to satisfy more than one of the following. --- Verify an OPERABLE SRM detector is located in: a. The fueled region ; b. The core quadrant where CORE ALTERATIONS are being performed, when the associated SRM is included in the fueled region ; and c. A core quadrant adjacent to where CORE ALTERATIONS are being performed, when the associated SRM is included in the fueled region.	12 hours	4 4
DOC M.6	SR 3.3.1.2.3	Perform CHANNEL CHECK.	24 hours	
3.3.B.4, 3.10.B.2, 4.10.B, DOC M.6	SR 3.3.1.2.4	---NOTE--- Not required to be met with less than or equal to ~~four~~ two fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant. --- Verify count rate is ~~:~~ ~~a.~~ ≥ 3.0 cps with a signal to noise ratio ≥ ~~[2:1]~~ 3:1 ~~or~~ ~~b. ≥ [0.7] cps with a signal to noise ratio ≥ [20:1].~~	12 hours during CORE ALTERATIONS 1 AND 24 hours	2 3
4.10.B	SR 3.3.1.2.5	INSERT 1 Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio.	7 days	5 1

5

INSERT 1

-----------------------------NOTE-----------------------------
The determination of signal to noise ratio is not required to be met with less than or equal to two fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant.

CTS

SURVEILLANCE REQUIREMENTS (continued)

		SURVEILLANCE	FREQUENCY	
DOC M.4, DOC M.6	SR 3.3.1.2.6	-----NOTE----- Not required to be performed until 12 hours after IRMs on Range 2 or below. ----- Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio.	31 days	1
DOC M.4, DOC M.6	SR 3.3.1.2.7	-----NOTES----- 1. Neutron detectors are excluded. 2. Not required to be performed until 12 hours after IRMs on Range 2 or below. ----- Perform CHANNEL CALIBRATION.	24 ~~[18]~~ months	1

SRM Instrumentation
3.3.1.2

CTS

Table 3.3.1.2-1 (page 1 of 1)
Source Range Monitor Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	
3.3.B.4	1. Source Range Monitor	2[(a)]	[3]	SR 3.3.1.2.1 SR 3.3.1.2.4 SR 3.3.1.2.6 SR 3.3.1.2.7	1
DOC M.6		3, 4	2	SR 3.3.1.2.3 SR 3.3.1.2.4 SR 3.3.1.2.6 SR 3.3.1.2.7	
3.10.B, 3.3.B.4		5	2[(b), (c)]	SR 3.3.1.2.1 SR 3.3.1.2.2 SR 3.3.1.2.4 SR 3.3.1.2.5 SR 3.3.1.2.7	

DOC M.7 (a) With IRMs on Range 2 or below.

3.10.B (b) Only one SRM channel is required to be OPERABLE during spiral offload or reload when the fueled region includes only that SRM detector.

3.10.B.1 (c) Special movable detectors may be used in place of SRMs if connected to normal SRM circuits.

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.1.2, SOURCE RANGE MONITOR (SRM) INSTRUMENTATION

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. Monticello is not licensed with the option for utilizing a lower count rate. Therefore, the requirement in ISTS SR 3.3.1.2.4.b is not included in the Monticello ITS and ITS SR 3.3.1.2.4 has been reformatted consistent with other places in the ITS.

4. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

5. A new Note has been added to ISTS SR 3.3.1.2.5 to state that the determination of the signal to noise ratio is not required to be met with less than or equal to two fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant. When starting to load fuel from the defueled condition, ISTS SR 3.3.1.2.5 must be current prior to the start of fuel load. However, with no fuel in the core, a signal to noise ratio cannot be determined. Therefore, this Note has been added similar to the Note in ISTS SR 3.3.1.2.4, which is included for the same reason as the proposed Mote. This change is consistent with the most recently approved BWR ITS conversions (i.e., FitzPatrick, LaSalle Units 1 and 2, Quad Cities Units 1 and 2, and Dresden Units 2 and 3) and proposed TSTF-455.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION

B 3.3.1.2 Source Range Monitor (SRM) Instrumentation

BASES

BACKGROUND

The SRMs provide the operator with information relative to the neutron flux level at very low flux levels in the core. As such, the SRM indication is used by the operator to monitor the approach to criticality and determine when criticality is achieved. The SRMs are maintained fully inserted until the count rate is greater than a minimum allowed count rate (a control rod block is set at this condition). After SRM to intermediate range monitor (IRM) overlap is demonstrated (as required by SR 3.3.1.1.1), the SRMs are normally fully withdrawn from the core. 5

The SRM subsystem of the Neutron Monitoring System (NMS) consists of four channels. Each of the SRM channels can be bypassed, but only one at any given time, by the operation of a bypass switch. Each channel includes one detector that can be physically positioned in the core. Each detector assembly consists of a miniature fission chamber with associated cabling, signal conditioning equipment, and electronics associated with the various SRM functions. The signal conditioning equipment converts the current pulses from the fission chamber to analog DC currents that correspond to the count rate. Each channel also includes indication, alarm, and control rod blocks. However, this LCO specifies OPERABILITY requirements only for the monitoring and indication functions of the SRMs.

During refueling, shutdown, and low power operations, the primary indication of neutron flux levels is provided by the SRMs or special movable detectors connected to the normal SRM circuits. The SRMs provide monitoring of reactivity changes during fuel or control rod movement and give the control room operator early indication of unexpected subcritical multiplication that could be indicative of an approach to criticality.

APPLICABLE SAFETY ANALYSES

Prevention and mitigation of prompt reactivity excursions during refueling and low power operation is provided by LCO 3.9.1, "Refueling Equipment Interlocks," LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," the IRM Neutron High
Flux - High and Average Power Range Monitor (APRM) Neutron Flux - 5
High, Setdown Functions, and LCO 3.3.2.1, "Control Rod Block Instrumentation."

U The SRMs have no safety function and are not assumed to function during any FSAR design basis accident or transient analysis. However, 1
the SRMs provide the only on scale monitoring of neutron flux levels during startup and refueling. Therefore, they are being retained in Technical Specifications.

BASES

LCO

During startup in MODE 2, three of the four SRM channels are required to be OPERABLE to monitor the reactor flux level prior to and during control rod withdrawal, subcritical multiplication and reactor criticality, and neutron flux level and reactor period until the flux level is sufficient to maintain the IRM on Range 3 or above. All but one of the channels are required in order to provide a representation of the overall core response during those periods when reactivity changes are occurring throughout the core.

In MODES 3 and 4, with the reactor shut down, two SRM channels provide redundant monitoring of flux levels in the core.

In MODE 5, during a spiral offload or reload, an SRM outside the fueled region will no longer be required to be OPERABLE, since it is not capable of monitoring neutron flux in the fueled region of the core. Thus, CORE ALTERATIONS are allowed in a quadrant with no OPERABLE SRM in an adjacent quadrant as provided the Table 3.3.1.2-1, footnote (b), (2) requirement that the bundles being spiral reloaded or spiral offloaded are all in a single fueled region containing at least one OPERABLE SRM is met. Spiral reloading and offloading encompass reloading or offloading a cell on the edge of a continuous fueled region (the cell can be reloaded or offloaded in any sequence).

In nonspiral routine operations, two SRMs are required to be OPERABLE to provide redundant monitoring of reactivity changes occurring in the reactor core. Because of the local nature of reactivity changes during refueling, adequate coverage is provided by requiring one SRM to be OPERABLE in the quadrant of the reactor core where CORE ALTERATIONS are being performed, and the other SRM to be OPERABLE in an adjacent quadrant containing fuel. These requirements ensure that the reactivity of the core will be continuously monitored during CORE ALTERATIONS.

in MODE 5

Special movable detectors, according to footnote (c) of Table 3.3.1.2-1, may be used ~~during CORE ALTERATIONS~~ in place of the normal SRM (3) nuclear detectors. These special detectors must be connected to the normal SRM circuits in the NMS, such that the applicable neutron flux indication can be generated. These special detectors provide more flexibility in monitoring reactivity changes during fuel loading, since they can be positioned anywhere within the core during refueling. They must still meet the location requirements of SR 3.3.1.2.2 and all other required SRs for SRMs.

For an SRM channel to be considered OPERABLE, it must be providing neutron flux monitoring indication.

and must be inserted to the normal operating level (1)

BASES

APPLICABILITY

The SRMs are required to be OPERABLE in MODE~~S~~ 2~~,~~ and MODES 3, 4, and 5 prior to the IRMs being on scale on Range 3 to provide for neutron monitoring. 3
In MODE 1, the APRMs provide adequate monitoring of reactivity changes in the core; therefore, the SRMs are not required. In MODE 2, with IRMs on Range 3 or above, the IRMs provide adequate monitoring and the SRMs are not required.

ACTIONS

A.1 and B.1

In MODE 2, with the IRMs on Range 2 or below, SRMs provide the means of monitoring core reactivity and criticality. With any number of the required SRMs inoperable, the ability to monitor neutron flux is degraded. Therefore, a limited time is allowed to restore the inoperable channels to OPERABLE status.

Provided at least one SRM remains OPERABLE, Required Action A.1 allows 4 hours to restore the required SRMs to OPERABLE status. This time is reasonable because there is adequate capability remaining to monitor the core, there is limited risk of an event during this time, and there is sufficient time to take corrective actions to restore the required SRMs to OPERABLE status or to establish alternate IRM monitoring capability. During this time, control rod withdrawal and power increase is not precluded by this Required Action. Having the ability to monitor the core with at least one SRM, proceeding to IRM Range 3 or greater with 5
overlap verified ~~by~~ SR 3.3.1.1.1, and thereby exiting the Applicability of this LCO, is acceptable for ensuring adequate core monitoring and allowing continued operation.

With three required SRMs inoperable, Required Action B.1 allows no positive changes in reactivity (control rod withdrawal must be immediately suspended) due to inability to monitor the changes. Required Action A.1 still applies and allows 4 hours to restore monitoring capability prior to requiring control rod insertion. This allowance is based on the limited risk of an event during this time, provided that no control rod withdrawals are allowed, and the desire to concentrate efforts on repair, rather than to immediately shut down, with no SRMs OPERABLE.

C.1

In MODE 2~~,~~ with the IRMs on Range 2 or below if the required number of SRMs is not restored to 3
OPERABLE status within the allowed Completion Time, the reactor shall be placed in MODE 3. With all control rods fully inserted, the core is in its least reactive state with the most margin to criticality. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full ~~power~~ conditions in an orderly 3
manner and without challenging plant systems.

BASES

ACTIONS (continued)

D.1 and D.2

With one or more required SRMs inoperable in MODE 3 or 4, the neutron flux monitoring capability is degraded or nonexistent. The requirement to fully insert all insertable control rods ensures that the reactor will be at its minimum reactivity level while no neutron monitoring capability is available. Placing the reactor mode switch in the shutdown position prevents subsequent control rod withdrawal by maintaining a control rod block. The allowed Completion Time of 1 hour is sufficient to accomplish the Required Action, and takes into account the low probability of an event requiring the SRM occurring during this interval.

E.1 and E.2

With one or more required SRM channels inoperable in MODE 5, the ability to detect local reactivity changes in the core during refueling is degraded. ~~CORE ALTERATIONS must be immediately suspended and~~ action must be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Suspending CORE ALTERATIONS prevents the two most probable causes of reactivity changes, fuel loading and control rod withdrawal, from occurring. Inserting all insertable control rods ensures that the reactor will be at its minimum reactivity given that fuel is present in the core. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe, conservative position. 3

Action (once required to be initiated) to insert control rods must continue until all insertable rods in core cells containing one or more fuel assemblies are inserted.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, ~~T~~the SRs for each SRM Applicable MODE or other specified conditions are found in the SRs column of Table 3.3.1.2-1. 4

SR 3.3.1.2.1 and SR 3.3.1.2.3

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on another channel. It is based on the assumption that instrument channels monitoring the same parameter should read

BASES

SURVEILLANCE REQUIREMENTS (continued)

approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency of once every 12 hours for SR 3.3.1.2.1 is based on operating experience that demonstrates channel failure is rare. While in MODES 3 and 4, reactivity changes are not expected; therefore, the 12 hour Frequency is relaxed to 24 hours for SR 3.3.1.2.3. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.2.2

To provide adequate coverage of potential reactivity changes in the core, one SRM is required to be OPERABLE in the quadrant where CORE ALTERATIONS are being performed, and the other OPERABLE SRM must be in an adjacent quadrant containing fuel. Note 1 states that the SR is required to be met only during CORE ALTERATIONS. It is not required to be met at other times in MODE 5 since core reactivity changes are not occurring. This Surveillance consists of a review of plant logs to ensure that SRMs required to be OPERABLE for given CORE ALTERATIONS are, in fact, OPERABLE. In the event that only one SRM is required to be OPERABLE, per Table 3.3.1.2-1, footnote (b), only the a. portion of this SR is required. Note 2 clarifies that more than one of the three requirements can be met by the same OPERABLE SRM. The 12 hour Frequency is based upon operating experience and supplements operational controls over refueling activities that include steps to ensure that the SRMs required by the LCO are in the proper quadrant.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2.4

with the detector full in

This Surveillance consists of a verification of the SRM instrument readout to ensure that the SRM reading is greater than a specified minimum count rate, which ensures that the detectors are indicating count rates indicative of neutron flux levels within the core. With few fuel assemblies loaded, the SRMs will not have a high enough count rate to satisfy the SR. Therefore, allowances are made for loading sufficient "source" material, in the form of irradiated fuel assemblies, to establish the minimum count rate. 3

two

To accomplish this, the SR is modified by a Note that states that the count rate is not required to be met on an SRM that has less than or equal to ~~four~~ fuel assemblies adjacent to the SRM and no other fuel assemblies are in the associated core quadrant. With ~~four~~ or less fuel assemblies loaded around each SRM and no other fuel assemblies in the associated core quadrant, even with a control rod withdrawn, the configuration will not be critical. 5

The Frequency is based upon channel redundancy and other information available in the control room, and ensures that the required channels are frequently monitored while core reactivity changes are occurring. When no reactivity changes are in progress, the Frequency is relaxed from 12 hours to 24 hours.

SR 3.3.1.2.5 and SR 3.3.1.2.6

Performance of a CHANNEL FUNCTIONAL TEST demonstrates the associated channel will function properly. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. SR 3.3.1.2.5 is required in MODE 5, and the 7 day Frequency ensures that the channels are OPERABLE while core reactivity changes could be in progress. This Frequency is reasonable, based on operating experience and on other Surveillances (such as a CHANNEL CHECK), that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2.6 is required in MODE 2 with IRMs on Range 2 or below, and in MODES 3 and 4. Since core reactivity changes do not normally take place, the Frequency is has been extended from 7 days to 31 days. The 31 day Frequency is based on operating experience and on other Surveillances (such as CHANNEL CHECK) that ensure proper functioning between CHANNEL FUNCTIONAL TESTS. 4

Verification of the signal to noise ratio also ensures that the detectors are inserted to an acceptable operating level. In a fully withdrawn condition, the detectors are sufficiently removed from the fueled region of the core to essentially eliminate neutrons from reaching the detector. Any count rate obtained while the detectors are fully withdrawn is assumed to be "noise" only.

INSERT 1 The Note to SR 3.3.1.2.6 the Surveillance allows the Surveillance to be delayed until entry into the specified condition of the Applicability (THERMAL POWER decreased to IRM Range 2 or below). The SR must be performed within 12 hours after IRMs are on Range 2 or below. The allowance to enter the Applicability with the 31 day Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. In this event, the 12 hour Frequency is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances. 5 3

SR 3.3.1.2.7

Performance of a CHANNEL CALIBRATION at a Frequency of 24 18 months verifies the performance of the SRM detectors and associated circuitry. The Frequency considers the plant conditions required to perform the test, the ease of performing the test, and the likelihood of a change in the system or component status. This SR is modified by two Notes. Note 1 excludes The neutron detectors are excluded from the CHANNEL CALIBRATION because they cannot readily be adjusted. The detectors are fission chambers that are designed to have a relatively constant sensitivity over the range and with an accuracy specified for a fixed useful life. 5 3 4

5

INSERT 1

With few fuel assemblies loaded, the SRMs will not have a high enough count rate to determine the signal to noise ratio. Therefore, allowances are made for loading sufficient source material, in the form of irradiated fuel assemblies, to establish the conditions necessary to determine signal to noise ratio. To accomplish this, SR 3.3.1.2.5 is modified by a Note that states that the determination of signal to noise ratio is not required to be met on an SRM that has less than or equal to two fuel assemblies adjacent to the SRM and no other fuel assemblies are in the associated core quadrant. With two or less fuel assemblies loaded around each SRM and no other fuel assemblies in the associated quadrant, even with the control rod withdrawn the configuration will not be critical.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Note 2 to the Surveillance allows the Surveillance to be delayed until entry into the specified condition of the Applicability. The SR must be performed in MODE 2 within 12 hours of entering MODE 2 with IRMs on Range 2 or below. The allowance to enter the Applicability with the 24 18 month Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. In this event, the 12 hour Frequency is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances. 5

REFERENCES None.

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.1.2 BASES, SOURCE RANGE MONITOR (SRM) INSTRUMENTATION

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.
2. Typographical/Grammatical error corrected.
3. Changes have been made to reflect the actual Specification requirements.
4. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.
5. Changes have been made to reflect those changes made to the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.1.2, SOURCE RANGE MONITOR (SRM) INSTRUMENTATION

There are no specific NSHC discussions for this Specification.

ATTACHMENT 3

ITS 3.3.2.1, Control Rod Block Instrumentation

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3.2 PROTECTIVE INSTRUMENTATION	4.2 PROTECTIVE INSTRUMENTATION
Applicability: Applies to the plant instrumentation which performs a protective function.	Applicability: Applies to the surveillance requirements of the instrumentation that performs a protective function.
Objective: To assure the operability of protective instrumentation.	Objective: To specify the type and frequency of surveillance to be applied to protective instrumentation.
Specification:	Specification:
A. Primary Containment Isolation Functions When primary containment integrity is required, the limiting conditions of operation for the instrumentation that initiates primary containment isolation are given in Table 3.2.1.	The instrumentation to be functionally tested and calibrated and the frequency of the tests is given in Table 4.2.1.

See ITS 3.3.6.1

Note 1 to Surveillance Requirements

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A.1

ITS | ITS

ITS	3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS	
	B. Emergency Core Cooling Subsystems Actuation When irradiated fuel is in the reactor vessel and the reactor water temperature is above 212°F, the limiting conditions for operation for the instrumentation which initiates the emergency core cooling subsystems are given in Table 3.2.2.	See ITS 3.3.5.1	
3.3.2.1	C. Control Rod Block ~~Actuation~~ Instrumentation	C. Control Rod Block ~~Actuation~~ Instrumentation	
	~~1. SRM, IRM, APRM and Scram Discharge Volume Rod Blocks~~ ~~The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.3.~~ R.1	~~During operation requiring RBM operability when only one channel is operable, an instrument functional test of the operable RBM shall be performed within 24 hours prior to withdrawal of control rod(s).~~	L.2
Table 3.3.2.1-1 Function 1	2. Rod Block Monitor (RBM) a. When core thermal power is greater than or equal to 30% of rated and MCPR is below the limits specified in the Core Operating Limits Report, either:	Add proposed Surveillance Requirements Note 2	L.3
	(1) Both RBM channels shall be operable, or	Add proposed Required Action A.1	L.1
ACTION A	(2) With one RBM channel inoperable, control rod withdrawal shall be blocked within ~~24~~ 1 hour hours, or		
ACTION B	(3) With both RBM channels inoperable, control rod withdrawal shall be blocked ~~immediately~~ 1 hour		

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ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. Rod Block Monitor (RBM) (continued)	
b. RBM ~~Setpoints~~ Allowable Values for control rod block are given in Table 3.2.3. The upscale LTSP shall be applied above 30% and up to 65% of rated thermal power. The upscale ITSP shall be applied at and above 65% and up to 85% of rated thermal power. The upscale HTSP shall be applied at and above 85% of rated thermal power. ~~The RBM Bypass time delay shall be less than or equal to 2.0 seconds.~~	
D. Other Instrumentation Whenever the reactor is in the RUN Mode, the limiting conditions for operation for the instrumentation listed in Table 3.2.8 shall be met.	

Table 3.3.2.1-1 Functions 1.a, 1.b, 1.c

footnote (a)

footnote (b)

footnote (c)

A.4

L.5

{See ITS 3.3.5.1 and ITS 3.3.5.2}

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Table 3.2.3
Instrumentation That Initiates Rod Block

Function	Trip Settings	Reactor Modes Which Function Must be Operable or Operating and Allowable Bypass Conditions** Refuel	Startup	Run	Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels per Trip System	Required Conditions *
1. SRM							
a. Upscale	≤5x10⁵ cps	X	X(d)		2	1(Note 1, 3, 6)	A or B or C
b. Detector not fully inserted		X(a)	X(a)		2	1 (Note 1, 3, 6)	A or B or C
2. IRM							
a. Downscale	≥3/125 full scale	X(b)	X(b)		4	2 (Note 1, 4, 6)	A or B or C
b. Upscale	≤108/125 full scale	X	X		4	2(Note 1, 4, 6)	A or B or C
3. APRM							
a. Upscale							
(1) TLO Flow Biased	≤0.66W + 53.6% (Note 2)			X	3	1(Note 1, 6, 7)	D or E
(2) SLO Flow Biased	≤0.66(W-5.4)+53.6% (Note 2)						
(3) High Flow Clamp	≤108%						
b. Downscale	≥3/125 full scale			X	3	1(Note 1, 6, 7)	D or E

R.1

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A.1

ITS

A.4

Table 3.3.2.1-1

Allowable Value

R.1 LA.1 R.1

Table 3.2.3 Continued Instrumentation That Initiates Rod Block							
Function	Trip Settings	Reactor Modes in Which Function Must be Operable or Operating and Allowable Bypass Conditions (5) Refuel	Startup	Run	Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels per Trip System	Required Conditions (*)
4. RBM (ITS 1)							
a. Upscale (power referenced). (ITS 1.a, 1.b, 1.c)	(Note 8)	See Section 3.2.C.2			1	See Section 3.2.C.2 (note 5)	See Section 3.2.C.2
b. Downscale (ITS 1.e)	≥ ~~94~~ 93.6/125 of full scale	See Section 3.2.C.2			1	See Section 3.2.C.2 (note 5)	See Section 3.2.C.2
5. Scram Discharge Volume Water Level High							
a. East	≤40 gal		X	X	1	1 (note 6)	B and D, or A
b. West	≤40 gal		X	X	1	1 (note 6)	B and D, or A

L.6 LA.2 R.1

Add proposed Table 3.3.2.1-1 Function 1.d — M.1

Add proposed Table 3.3.2.1-1 Function 3 and ACTION E — M.2

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Table 3.2.3 - Continued
Instrumentation That Initiates Rod Block

Table 3.3.2.1-1

Notes:

~~(1) There shall be two operable or operating trip systems for each function. If the minimum number of operable or operating instrument channels cannot be met for one of the two trip systems, this condition may exist up to seven days provided that during this time the operable system is functionally tested immediately and daily thereafter.~~

~~(2) "W" is the percent of drive flow required to produce a rated core flow of 57.6 x 10^6 lb/hr~~

~~(3) Only one of the four SRM channels may be bypassed.~~

~~(4) There must be at least one operable or operating IRM channel monitoring each core quadrant.~~

R.1

~~(5) An RBM channel will be considered inoperable if there are less than half the total number of normal inputs.~~

LA.2

~~(6) Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied actions shall be initiated to:~~

~~(a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition or~~

~~(b) Place the plant under the specified required conditions using normal operating procedures.~~

~~(7) There must be a total of at least 4 operable or operating APRM channels~~

R.1

Table 3.3.2.1-1
Functions 1.a, 1.b, 1.c and Footnotes (a), (b), (c)

(8) There are 3 upscale trip levels. Only one is applied over a specified operating core thermal power range. All RBM trips are automatically bypassed below 30% thermal power. ~~Trip settings~~ Allowable Values are provided in the Core Operating Limits Report.

A.4

Table 3.2.3 Continued
Instrumentation That Initiates Rod Block

Notes:

*Required conditions when minimum conditions for operation are not satisfied.

A. Reactor in Shutdown mode.

B. No rod withdrawals permitted while in Refuel or Startup mode.

C. Reactor in Run mode.

D. No rod withdrawals permitted while in the Run mode.

E. Power on IRM range or below and reactor in Startup, Refuel, or Shutdown mode.

**Allowable Bypass Conditions

a. SRM Detector-not-fully-inserted rod block may be bypassed when the SRM channel count rate is ≥ 100 cps or when all IRM range switches are above Position 2.

b. IRM Downscale rod block may be bypassed when the IRM range switch is in the lowest range position.

c. (deleted)

d. SRM Upscale block may be bypassed when associated IRM range switches are above Postion 8.

R.1

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Table 3.3.2.1-1

Table 4.2.1
Minimum Test and Calibration Frequency for Core Cooling, Rod Block and Isolation Instrumentation

SR 3.3.2.1.1 | SR 3.3.2.1.4, SR 3.3.2.1.5 | A.2

ITS	Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
	ECCS INSTRUMENTATION			
	1. Reactor Low-Low Water Level	Once/3 months (Note 5)	Every Operating Cycle - Transmitter Once/3 months -Trip Unit	Once/12 hours
	2. Drywell High Pressure	Once/3 months	Once/3 months	None
	3. Reactor Low Pressure (Pump Start)	Once/3 months	Once/3 months	None
	4. Reactor Low Pressure (Valve Permissive)	Once/3 months	Once/3 months	None
	5. Undervoltage Emergency Bus	Refueling Outage	Refueling Outage	None
	6. Low Pressure Core Cooling Pumps Discharge Pressure Interlock	Once/3 months	Once/3 months	None
	7. Loss of Auxiliary Power	Refueling Outage	Refueling Outage	None
	8. Condensate Storage Tank Level	Refueling Outage	Refueling Outage	None
	9. Reactor High Water Level	Once/3 months (Note 5)	Every Operating Cycle - Transmitter Every 3 months - Trip Unit	Once/12 hours
1	ROD BLOCKS			
	~~1. APRM Downscale~~	~~Once/3 months (Note 5)~~	~~Once/3 months~~	~~None~~
	~~2. APRM Flow Variable~~	~~Once/3 months (Note 5)~~	~~Once/3 months~~	~~None~~
	~~3. IRM Upscale~~	~~Notes (2,5)~~	~~Note 2~~	~~Note 2~~
	~~4. IRM Downscale~~	~~Notes (2,5)~~	~~Note 2~~	~~Note 2~~
1.a, 1.b, 1.c	5. RBM Upscale	Once/3 months ~~(Note 5)~~	Once/3 months	None
1.e	6. RBM Downscale	Once/3 months ~~(Note 5)~~	Once/3 months	None
	~~7. SRM Upscale~~	~~Notes (2,5)~~	~~Note 2~~	~~Note 2~~
	~~8. SRM Detector Not-Full-In Position~~	~~Notes (2,8)~~	~~Note 2~~	~~None~~
	~~9. Scram Discharge Volume-High Level~~	~~Once/3 months~~	~~Refueling Outage~~	~~None~~
	MAIN STEAM LINE (GROUP 1) ISOLATION			
	1. Steam Tunnel High Temperature	Refueling Outage Once/3 months	Refueling Outage	None
	2. Steam Line High Flow	Once/3 months	Once/3 Months	Once/12 hours

See ITS 3.3.5.1

R.1

A.3

R.1

See ITS 3.3.6.1

A.5

Add proposed SR 3.3.2.1.1 for Function 1.d — M.1

Amendment No. ~~2~~, ~~10~~, ~~37~~, ~~39~~, ~~63~~, ~~66~~, ~~81~~, ~~103~~, 104

Add proposed SR 3.3.2.1.7 — M.2

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A.1

Table 4.2.1 Continued
Minimum Test and Calibration Frequency for Core Cooling,
Rod Block and Isolation Instrumentation

NOTES:

(1) (Deleted)

(2) Calibrate prior to normal shutdown and start-up and thereafter check once per 12 hours and test once per week until no longer required. Calibration of this instrument prior to normal shutdown means adjustment of channel trips so that they correspond, within acceptable range and accuracy, to a simulated signal injected into the instrument (not primary sensor). In addition, IRM gain adjustment will be performed, as necessary, in the APRM/IRM overlap region. — R.1

(3) Functional tests, calibrations and sensor checks are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status. — A.2

(4) Whenever fuel handling is in process, a sensor check shall be performed once per 12 hours. — See ITS 3.3.6.2

(5) A functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and/or initiating action. — A.3

(6) (Deleted)

(7) (Deleted)

(8) Once/shutdown if not tested during previous 3 month period. — See ITS 3.3.6.3

(9) Testing of the SRM Not-Full-In rod block is not required if the SRM detectors are secured in the full-in position. — R.1

(10) Uses contacts from scram system. Tested and calibrated in accordance with Tables 4.1.1 and 4.1.2. — See ITS 3.3.6.1 and ITS 3.3.6.2

(11) Uses contacts from Group 1 Isolation logic. Tested and calibrated in accordance with Group 1 Low Low Water Level Instrumentation.

(12) Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel.

See ITS 3.3.6.1

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A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	(b) when the rod is withdrawn the first time subsequent to each refueling outage, observe discernible response of the nuclear instrumentation. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is critical shall be performed to observe nuclear instrumentation response. — See ITS 3.1.3
2. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all operable control rods are fully inserted and Specification 3.3.A.1 is met.	2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded. — See CTS 3/4.3.B.2
See ITS 3.1.6 — 3.(a) Control rod withdrawal sequences shall be established so that the maximum calculated reactivity that could be added by dropout of any increment of any one control blade will not make the core more than 1.3% Δk supercritical.	3.(a) To consider the rod worth minimizer operable, the following steps must be performed: SR 3.3.2.1.8 (i) The control rod withdrawal sequence for the rod worth minimizer computer shall be verified as correct. SR 3.3.2.1.2, SR 3.3.2.1.3 ~~(ii) The rod worth minimizer computer on-line diagnostic test shall be successfully completed.~~ ~~(iii) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.~~ — LA.3

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ITS

ITS

Add proposed Note to SR 3.3.2.1.2 and SR 3.3.2.1.3 — M.5

every 92 days — M.5

ITS	3.0 LIMITING CONDITIONS FOR OPERATION	ITS	4.0 SURVEILLANCE REQUIREMENTS
Table 3.3.2.1-1, Function 2; ACTIONS C and D; Required Action C.2.1.1	(b) Whenever the reactor is in the startup or run mode below 10% rated thermal power, no control rods shall be moved unless the rod worth minimizer is operable or a second independent operator or engineer verifies that the operator at the reactor console is following the control rod program. The second operator may be used as a substitute for an inoperable rod worth minimizer during a startup only if the rod worth minimizer fails after withdrawal of at least twelve control rods.	SR 3.3.2.1.2, SR 3.3.2.1.3	(iv) The rod block function of the rod worth minimizer shall be verified ~~by attempting to withdraw an out-of-sequence control rod beyond the block point.~~
		ACTIONS C and D	(b) If the rod worth minimizer is inoperable while the reactor is in the startup or run mode below 10% rated thermal power and the second independent operator or engineer is being used, he shall verify that all rod positions are correct prior to commencing withdrawal or insertion of each rod group.
	4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.		4. Prior to control rod withdrawal for startup or during refueling verify that at least two source range channels have an observed count rate of at least three counts per second.

M.4 (LCO (b)); LA.3 (SR (iv)); M.4 (SR (b))

Add proposed Required Action C.2.1.2 — L.4

{ See ITS 3.3.1.2 }

Add proposed SR 3.3.2.1.6 — M.3

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ITS

Table 3.3.2.1-1
Function 3

Y. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. In this condition, a reactor scram is initiated and a rod block is inserted directly from the mode switch. The scram can be reset after a short time delay.

See ITS 1.0

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.
2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

Z. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

See ITS 1.0

AA. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling, also referred to as partial nucleate boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

AB. Pressure Boundary Leakage - Pressure boundary leakage shall be leakage through a non-isolable fault in the reactor coolant system pressure boundary.

AC. Identified Leakage - Identified leakage shall be:

1. Leakage into the drywell, such as that from pump seals or valve packing leaks, that is captured and conducted to a sump or collecting tank, or
2. Leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be Pressure Boundary Leakage.

AD. Unidentified Leakage - All leakage into the drywell that is not Identified Leakage.

AE. Total Leakage - Sum of the Identified and Unidentified Leakage.

AF. through AH. (Deleted)

AI. Purging - Purging is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

AJ. Venting - Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required.

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DISCUSSION OF CHANGES
ITS 3.3.2.1, CONTROL ROD BLOCK INSTRUMENTATION

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS Table 4.2.1 Note (3) states that functional tests, calibrations, and sensor checks are not required when these instruments are not required to be OPERABLE or are tripped. In addition, the Note states that if tests are missed, they shall be performed prior to returning the systems to an OPERABLE status. These explicit requirements are not retained in ITS 3.3.2.1. This changes the CTS by not including these explicit requirements.

The purpose of this Note is to provide guidance on when Surveillances are required to be met and performed. This explicit Note is not needed in ITS 3.3.2.1 since these allowances are included in ITS SR 3.0.1. ITS SR 3.0.1 states that SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR, and failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. SR 3.0.1 also states that SRs are not required to be performed on inoperable equipment. When equipment is declared inoperable, the Actions of this LCO require the equipment to be placed in the trip condition. In this condition, the equipment is still inoperable but has accomplished the required safety function. Therefore the allowances in SR 3.0.1 and the associated actions provide adequate guidance with respect to when the associated surveillances are required to be performed and this explicit requirement is not retained. This change is administrative because it does not result in a technical change to the CTS.

A.3 CTS Table 4.2.1 Note (5) states that a functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and or initiating action. These explicit requirements are not retained in ITS 3.3.2.1. This changes the CTS by not including these explicit requirements.

The purpose of CTS Table 4.2.1 Note (5) is to provide guidance on how to perform an instrument functional test of the RBM channels. This explicit Note is not needed in ITS 3.3.2.1 since the requirements for the CHANNEL FUNCTIONAL TEST are included in ITS 1.0, "Definitions." ITS 1.0 states that a CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. Therefore, the ITS 1.0 definition provides adequate guidance with respect to performance requirements of a CHANNEL FUNCTIONAL TEST and this explicit requirement is not retained. This change is administrative because it does not result in a technical change to the CTS.

DISCUSSION OF CHANGES
ITS 3.3.2.1, CONTROL ROD BLOCK INSTRUMENTATION

A.4 CTS 3.2.C.2.b states that the RBM "Setpoints" for the control rod block are given in Table 3.2.3. CTS Table 3.2.3 specifies the "Trip Settings" for each RBM Function. ITS LCO 3.3.2.1 requires the control rod block instrumentations for each Function in Table 3.3.2.1-1 to be OPERABLE and ITS Table 3.3.2.1-1 specifies the "Allowable Value" for each Function. This changes the CTS by replacing the terms "Setpoints" and "Trip Settings" with "Allowable Value."

The purpose of the "Trip Settings" in CTS Table 3.2.3 is to define the OPERABILITY limits for the RBM instrumentation Functions. Therefore, the use of the terms "Setpoints" and "Trip Settings" in the CTS is the same as the use of the term "Allowable Value" in the ITS. This proposed change does not modify the actual "Trip Settings" specified in CTS Table 3.2.3 for the RBM Functions. Any changes to the actual "Trip Settings" (i.e., changing the value for OPERABILITY) are discussed in DOC L.6. This change is designated as administrative change and is acceptable because it does not result in any technical changes to the CTS.

A.5 This change to CTS Table 4.2.1 is provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the USNRC for approval in NMC letter L-MT-04-036, from Thomas J. Palmisano (NMC) to USNRC, dated June 30, 2004. As such, this change is administrative.

MORE RESTRICTIVE CHANGES

M.1 CTS Table 3.2.3 does not include any requirements for the Rod Block Monitor "Inop" Function. ITS Table 3.3.2.1-1, Function 1.d, requires the Rod Block Monitor-Inop Function be OPERABLE and specifies performance of ITS SR 3.3.2.1.1, a CHANNEL FUNCTIONAL TEST, every 92 days. This changes the CTS by adding requirements for a RBM Function that was not previously required.

The purpose of CTS Table 3.2.3, in part, is to ensure the Rod Block Monitor is capable of performing its function. Similarly, ITS Table 3.3.2.1-1 Function 1 includes those Rod Block Monitor Functions required by CTS Table 3.2.3, but also specifies the additional Rod Block Monitor - Inop Function, Function 1.d, and the applicable Surveillance Requirement, ITS SR 3.3.2.1.1. This change is acceptable because it ensures a rod block is provided if the minimum number of LPRM inputs is not available to the associated Rod Block Monitor channel. In addition, ITS SR 3.3.2.1.1 requires a CHANNEL FUNCTIONAL TEST for each RBM channel to ensure the channel will perform its intended function when it is required to be OPERABLE. The proposed Frequency of 92 days for ITS SR 3.3.2.1.1 is based on the reliability analysis provided in NEDC-30851-P-A. Therefore, the addition of the Rod Block Monitor - Inop Function, its associated CHANNEL FUNCTIONAL TEST SR 3.3.2.1.1 and the 92 day Surveillance interval will help to ensure that the local flux is adequately monitored during control rod withdrawal by promptly identifying to the operator the inoperability of the Rod Block Monitor as a consequence of certain component failures. This change is more restrictive because the ITS specifies requirements for a Function and associated Surveillance not currently required by the CTS.

DISCUSSION OF CHANGES
ITS 3.3.2.1, CONTROL ROD BLOCK INSTRUMENTATION

M.2 CTS Table 3.2.3 does not include any requirements for the "Reactor Mode Switch - Shutdown Position" Function. ITS 3.3.2.1 includes the LCO, Required Actions, and Surveillance Requirement for the Reactor Mode Switch - Shutdown Position Function consistent with the requirement of CTS 1.0.Y for a rod block to be inserted when the reactor mode switch is in the shutdown position. ITS Table 3.3.2.1-1 Function 3 requires two channels of the Reactor Mode Switch - Shutdown Position to be OPERABLE. ITS SR 3.3.2.1.7 requires performance of a CHANNEL FUNCTIONAL TEST every 24 months, and is modified by a Note which specifies the SR is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position. ITS 3.3.2.1 ACTION E addresses the requirements for an inoperable Reactor Mode Switch - Shutdown Position. This changes the CTS by adding a Reactor Mode Switch - Shutdown Position Function, associated Surveillance, and ACTION not previously required.

The purpose of CTS Table 3.2.3, in part, is to ensure that instrumentation that initiates rod blocks is OPERABLE. Similarly, ITS Table 3.3.2.1-1 includes, in part, those rod block Functions required by CTS Table 3.2.3 but also specifies the addition of Function 3, Reactor Mode Switch - Shutdown Position. ITS Table 3.3.2.1-1 Function 3 requires two channels of the Reactor Mode Switch - Shutdown Position to be OPERABLE. ITS SR 3.3.2.1.7 requires performance of a CHANNEL FUNCTIONAL TEST every 24 months, and is modified by a Note which specifies the SR is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position. ITS 3.3.2.1 ACTION E addresses the requirements for an inoperable Reactor Mode Switch - Shutdown Position channel. This change is acceptable because it ensures a rod block is provided when the reactor mode switch is placed in the shutdown position, thereby preventing inadvertent criticality as a result of control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. This change is more restrictive because the ITS specifies a Function, associated Surveillance, and Required Actions not currently required by the CTS.

M.3 ITS SR 3.3.2.1.6 requires verification that the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP every 24 months. This specific Surveillance is not required by CTS. This changes the CTS by adding a Surveillance Requirement that was not previously required.

The purpose of CTS 3.3.B.3.(b) is to specify, in part, the applicable conditions under which the RWM is required to be OPERABLE. ITS SR 3.3.2.1.6 performs a verification that the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP every 24 months. This change is acceptable because it ensures the RWM automatic bypass setpoint is periodically verified. The proposed Frequency of 24 months is based on engineering judgment considering the reliability of the components, and that indication of whether or not the RWM is bypassed is provided in the control room. This change is more restrictive because the ITS specifies Surveillance Requirements not currently required by the CTS.

M.4 CTS 3.3.B.3.(b) requires the RWM to be OPERABLE in the startup or run mode below 10% rated thermal power. ITS Table 3.3.2.1-1 Function 2 requires the

DISCUSSION OF CHANGES
ITS 3.3.2.1, CONTROL ROD BLOCK INSTRUMENTATION

RWM to be OPERABLE in MODES 1 and 2 when ≤ 10% RTP. This changes the CTS by requiring the RWM to be OPERABLE at exactly 10% RTP.

The purpose of the RWM is to enforce the BPWS to ensure that the initial conditions of the CRDA analysis are not violated. The assumptions of the analysis assume the RWM is OPERABLE when ≤ 10% RTP, since when > 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed 280 cal/gm fuel damage limit during a CRDA. Therefore, this change is acceptable. This change is more restrictive because the RWM is required OPERABLE under more conditions in the ITS than in the CTS.

M.5 CTS 4.3.B.3.(a)(iv) requires verifying the rod block function of the rod worth minimizer is OPERABLE whenever the reactor is in startup or run mode below 10% rated thermal power. However, no specific Frequency is provided. ITS 3.3.2.1 performs this verification using two Surveillance Requirements, each modified by a Note pertaining to the Applicability. ITS SR 3.3.2.1.2 requires performing a CHANNEL FUNCTIONAL TEST every 92 days, and is modified by a Note which specifies that this SR is not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. ITS SR 3.3.2.1.3 requires performing a CHANNEL FUNCTIONAL TEST every 92 days, and is modified by a Note which specifies that this SR is not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. This changes the CTS by specifying separate Surveillance Requirements at specific Surveillance Frequencies.

The purpose of CTS 4.3.B.3.(a)(iv) is to ensure the rod block function of the rod worth minimizer is OPERABLE whenever the reactor is in startup or run mode below 10% rated thermal power. Similarly, ITS SR 3.3.2.1.2 requires performing a CHANNEL FUNCTIONAL TEST every 92 days, and is modified by a Note which specifies that this SR is not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. In addition, ITS SR 3.3.2.1.3 requires performing a CHANNEL FUNCTIONAL TEST every 92 days, and is modified by a Note which specifies that this SR is not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. This change is acceptable because it retains the requirement to verify the rod block function of the rod worth minimizer and requires verification at a specified Frequency. In addition, this change by separating the requirement into two Surveillances with Notes of Applicability and specified conditions, establishes more specific requirements. This change is more restrictive because the ITS specifies Surveillance Frequencies not currently required by the CTS.

<u>RELOCATED SPECIFICATIONS</u>

R.1 CTS 3.2.C.1 and CTS Tables 3.2.3 and 4.2.1, in part, specify the limiting conditions of operation, associated Actions, and Surveillance Requirements for the Source Range Monitor (SRM), Intermediate Range Monitor (IRM), Average Power Range Monitor (APRM), and Scram Discharge Volume Rod Block Functions.

DISCUSSION OF CHANGES
ITS 3.3.2.1, CONTROL ROD BLOCK INSTRUMENTATION

The SRM, IRM, APRM, and Scram Discharge Volume rod blocks are intended to prevent rod withdrawal when plant conditions make such withdrawal imprudent. However, there are no safety analyses that depend on these rod blocks to prevent, mitigate, or establish initial conditions for a Design Basis Accident or transient. The evaluation summarized in NEDO-31466, determined that the loss of SRM, IRM, APRM, and Scram Discharge Volume rod blocks would be a non-significant risk contributor to core damage frequency and offsite releases. These Requirements do not meet the criteria for retention in the ITS; therefore, they will be retained in the Technical Requirements Manual (TRM).

This change is acceptable because CTS 3.2.C.1 and CTS Tables 3.2.3 and 4.2.1, in part, for SRM, IRM, APRM, and Scram Discharge Volume rod blocks, do not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The SRM, IRM, APRM, and Scram Discharge Volume rod blocks 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. The SRM, IRM, APRM, and Scram Discharge Volume rod blocks do not satisfy criterion 1.

2. The SRM, IRM, APRM, and Scram Discharge Volume rod block limits are not a process variable 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. The SRM, IRM, APRM, and Scram Discharge Volume rod blocks do not satisfy criterion 2.

3. The SRM, IRM, APRM, and Scram Discharge Volume rod blocks 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. The SRM, IRM, APRM, and Scram Discharge Volume rod blocks do not satisfy criterion 3.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (items 135, 137, 138 and 139) of NEDO-31466, the SRM, IRM, APRM, and Scram Discharge Volume rod blocks were found to be non-significant risk contributors to core damage frequency and offsite releases. Nuclear Management Company, LLC has reviewed this evaluation, considers it applicable to Monticello, and concurs with the assessment.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the SRM, IRM, APRM, and Scram Discharge Volume rod blocks LCO and associated Surveillances may be relocated out of the Technical Specifications. The SRM, IRM, APRM, and Scram Discharge Volume rod blocks will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the LCO did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

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ITS 3.3.2.1, CONTROL ROD BLOCK INSTRUMENTATION

REMOVED DETAIL CHANGES

LA.1 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.2.3 for Rod Block instrumentation Functions includes the column "Total No. of Instrument Channels per Trip System." ITS Table 3.3.2.1-1 does not retain this column. This changes the CTS by moving the information of the "Total No. of Instrument Channels per Trip System" column to the ITS Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the "REQUIRED CHANNELS." Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA.2 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.2.3 Note (5) specifies that the RBM channel is inoperable if there are less than half the total number of normal inputs. ITS Table 3.3.2.1-1 does not retain this information. This changes the CTS by moving the specific conditions of RBM OPERABILITY to the ITS Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the RBM – Inop Function to be OPERABLE (see DOC M.1). Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA.3 *(Type 3 – Removing Procedural Details for Meeting TS Requirements or Reporting Requirements)* CTS 4.3.B.3.(a)(ii) states that "The rod worth minimizer computer on-line diagnostic test shall be successfully completed." CTS 4.3.B.3.(a)(iii) states that "Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified." CTS 4.3.B.3.(a)(iv) specifies that the rod worth minimizer rod block function be verified "by attempting to withdrawal an out-of-sequence control rod beyond the block point." The ITS does not include these requirements. This changes the CTS by moving the specific details for performing rod worth minimizer testing to the ITS Bases.

DISCUSSION OF CHANGES
ITS 3.3.2.1, CONTROL ROD BLOCK INSTRUMENTATION

The removal of these details for performing Surveillance Requirements from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS SR 3.3.2.1.2 and ITS SR 3.3.2.1.3 still retain the requirement for performing a CHANNEL FUNCTIONAL TEST to verify the rod worth minimizer is OPERABLE. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is a less restrictive removal of detail change because procedural details for meeting Technical Specifications requirements are being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* CTS 3.2.C.2.a.(2) requires control rod withdrawal to be blocked within 24 hours if one channel is inoperable. CTS 3.2.C.2.a.(3) requires control rod withdrawal to be blocked immediately if two RBM channels are inoperable. ITS 3.3.2.1 Required Action A.1 allows 24 hours to restore one inoperable RBM channel. ITS 3.3.2.1 ACTION B allows 1 hour to place a RBM channel in trip if the Required Action and associated Completion Time of Condition A is not met, or if two RBM channels are inoperable. This changes the CTS by providing an additional 1 hour to evaluate and restore the inoperable RBM channels before requiring a channel to be placed in trip, thereby blocking control rod withdrawal.

The purpose of CTS 3.2.C.2.a.(2) and CTS 3.2.C.2.a.(3) is to prevent control rod withdrawal consistent with the inoperability of one or more RBM channels. Similarly, ITS 3.3.2.1 ACTIONS A and B establish the same restrictions but allow slightly more time before initiating a control rod block. ITS 3.3.2.1 ACTION A allows 24 hours to restore one inoperable RBM channel. If the Required Action and associated Completion Time is not met, ITS 3.3.2.1 Required Action B.1 allows 1 hour to place one RBM channel in trip. ITS 3.3.2.1 ACTION B also allows 1 hour to place one RBM channel in trip if both RBM channels are inoperable. These changes are acceptable because the allowance for 1 hour to place the RBM channel in trip allows the operator time to evaluate and repair any discovered inoperabilities. This change is less restrictive because less stringent Required Actions are being applied in ITS than were applied in CTS.

L.2 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.2.C requires performance of an instrument functional test of the OPERABLE RBM when one RBM channel is inoperable. ITS 3.3.2.1 does not include this Surveillance. This changes the CTS by deleting this Surveillance.

The purpose of the performance of an instrument functional test is to ensure the OPERABLE RBM will be able to meet its functional requirements. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and

DISCUSSION OF CHANGES
ITS 3.3.2.1, CONTROL ROD BLOCK INSTRUMENTATION

at a Frequency necessary to give confidence that the equipment can perform its assumed safety function. While the specified Surveillance has been deleted, other Surveillances are included which help to ensure the OPERABLE RBM channel will function as designed. ITS SR 3.3.2.1.1 requires performance of a CHANNEL FUNCTIONAL TEST every 92 days. This Surveillance is sufficient to ensure the OPERABLE RBM channel is acceptable to meet the design requirements. This change is less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L.3 *(Category 4 – Relaxation of Required Action)* CTS 3.2.C does not provide a delayed entry into associated Conditions and Required Actions if a RBM channel is inoperable for performance of required Surveillances. ITS 3.3.2.1 Surveillance Requirements Note 2 allows delayed entry into associated Conditions and Required Actions for up to 6 hours if a RBM channel is placed in an inoperable status for performance of required Surveillances provided the associated Function maintains rod block capability. This changes the CTS by providing a delay time to enter Conditions and Required Actions for a RBM channel placed in an inoperable status solely for performance of required Surveillances.

ITS 3.3.2.1 Surveillance Requirements Note 2 has been added to allow delayed entry into associated Conditions and Required Actions for up to 6 hours if a RBM channel is placed in an inoperable status for performance of required Surveillances provided the associated Function maintains rod block capability. This change is acceptable because it provides a reasonable time for performing tests and reduces the risk of error during testing. The 6 hour time is acceptable based on the average completion time of 3 to 4 hours for an individual test. In addition, this allowance is consistent with allowances in GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992. The logic design of RBM instrumentation is bounded by this reliability analysis and the conclusions of the analysis are applicable to the Monticello design. The result of the NRC review of this generic reliability analyses as it relates to Monticello is documented in the NRC Safety Evaluation Report (SER) for Amendment 103, dated December 23, 1998. The SER concluded that the generic reliability analysis is applicable to Monticello, and that Monticello meets all requirements of the NRC SER accepting the generic reliability analysis. This change is less restrictive because less stringent Required Actions are being applied in ITS than were applied in CTS.

L.4 *(Category 4 – Relaxation of Required Action)* CTS 3.3.B.3.(b) allows reactor startup to continue with the rod worth minimizer inoperable only if ≥ 12 control rods have already been withdrawn. ITS 3.3.2.1 Required Action C.2.1.2 allows startup to continue with the RWM inoperable and < 12 rods withdrawn if it is verified that a startup with the RWM inoperable has not been performed in the last 12 months. This changes the CTS by providing an additional allowance to continue rod withdrawal with a RWM inoperable.

The purpose of CTS 3.3.B.3.(b), in part, is to allow control rod withdrawal with the rod worth minimizer inoperable. Similarly, ITS 3.3.2.1 Required Actions C.2.1.1 and C.2.1.2 provide this allowance. Additionally ITS 3.3.2.1 Required Action C.2.1.2 allows startup with the RWM inoperable with < 12 rods withdrawn if it is immediately verified that startup with the RWM inoperable has not been performed in the last 12 months. This change is acceptable because the verification that a startup with the RWM inoperable has not been performed in the last 12 months is consistent with the conclusions in the NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987, that the Technical Specifications for the RWM require use of the RWM to an extent that would minimize substitution of a second operator and provide a strong incentive to maintain and improve that system. This change is less restrictive because less stringent Required Actions are being applied in ITS than were applied in CTS.

L.5 CTS 3.2.C.2.b states that the RBM bypass time delay must be less than or equal to 2.0 seconds. ITS 3.3.2.1 does not require the RBM bypass time delay to be OPERABLE. This changes the CTS by deleting the RBM bypass time delay requirements.

The purpose of CTS 3.2.C.2.b is to ensure that the RBM bypass time delay is within the assumed limit. After the RBM upscale trip exceeds the trip setpoint, the generated control rod block is allowed to be delayed for a short time (up to 2 seconds as allowed by CTS 3.2.C.2.b) by the RBM bypass time delay, prior to sending the control rod block signal to the associated Reactor Manual Control System rod block circuit. The safety analysis does not require a time delay, but only assumes the signal is delayed for up to 2.0 seconds. The RBM includes an electronic "dip" switch that bypasses the RBM bypass time delay feature. When the switch is placed in the bypass position, the RBM bypass time delay is effectively removed from the RBM circuitry; i.e., the time delay is set to zero seconds. Monticello performed a modification to the RBM System a few years ago (as part of the APRM, RBM, and Technical Specifications modification), and at that time set the "dip" switch to the bypass position as a permanent feature of the modification. Thus, the RBM control rod block signal is not currently being delayed. Therefore, the CTS allowance to have a RBM bypass time delay (set at up to 2.0 seconds) is not needed. This change is less restrictive because the LCO requirement for a RBM bypass time delay feature is being removed from the CTS.

L.6 *(Category 10 – Changing Instrumentation Allowable Values)* CTS Table 3.2.3 specifies the "Trip Settings" for the RBM instrumentation. The Trip Setting value of CTS Table 3.2.3 Function 4.b has been modified to reflect a new Allowable Value as indicated in ITS Table 3.3.2.1-1 Function 1.e. This changes the CTS by requiring the RBM Downscale instrumentation to be set consistent with the new "Allowable Value." The change in the term "Trip Settings" to "Allowable Value" is discussed in DOC A.4.

The purpose of the Allowable Values is to ensure the instruments function as assumed in the safety analyses. ITS 3.3.2.1 reflects Allowable Values consistent with the philosophy of General Electric ISTS, NUREG-1433. These Allowable Values have been established using the GE setpoint methodology guidance, as specified in the Monticello setpoint methodology. The analytic limits are derived from limiting values of the process parameters obtained from the safety analysis. The Allowable Value is derived from the design limit. The difference between the design limit and the Allowable Value allows for channel instrument accuracy, calibration accuracy, process measurement accuracy, and primary element accuracy. The margin between the Allowable Value and the nominal trip setpoint (NTSP) allows for instrument drift that might occur during the established surveillance period. Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift. These two verifications are statistical evaluations to provide additional assurance of the acceptability of the NTSP and may require changes to the NTSP. Use of these methods and verifications provides the assurance that if the setpoint is found conservative to the Allowable Value during surveillance testing, the instrumentation would have provided the required trip function by the time the process reached the design limit for the applicable events. Therefore, based on the above discussion, the inclusion of the Allowable Value as the OPERABILITY value in lieu of the Trip Setting is acceptable. This change is less restrictive because less stringent OPERABILITY values are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

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3.3 INSTRUMENTATION

3.2.C 3.3.2.1 Control Rod Block Instrumentation

See Table LCO 3.3.2.1 The control rod block instrumentation for each Function in Table 3.3.2.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2.1-1.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.2.C.2.a.(2)	A. One rod block monitor (RBM) channel inoperable.	A.1 Restore RBM channel to OPERABLE status.	24 hours
3.2.C.2.a.(2), 3.2.C.2.a.(3)	B. Required Action and associated Completion Time of Condition A not met. OR Two RBM channels inoperable.	B.1 Place one RBM channel in trip.	1 hour
3.3.B.3.(b), 4.3.B.3.(b)	C. Rod worth minimizer (RWM) inoperable during reactor startup.	C.1 Suspend control rod movement except by scram. OR C.2.1.1 Verify ≥ 12 rods withdrawn. OR	Immediately Immediately

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ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.3.B.3.(b)		C.2.1.2 Verify by administrative methods that startup with RWM inoperable has not been performed in the last $\boxed{\text{calendar year}}$. 12 months AND	Immediately (4)
		C.2.2 Verify movement of control rods is in compliance with banked position withdrawal sequence (BPWS) by a second licensed operator or other qualified member of the technical staff.	During control rod movement
4.3.B.3.(b)	D. RWM inoperable during reactor shutdown.	D.1 Verify movement of control rods is in compliance with BPWS by a second licensed operator or other qualified member of the technical staff.	During control rod movement
DOC M.2	E. One or more Reactor Mode Switch - Shutdown Position channels inoperable.	E.1 Suspend control rod withdrawal. AND	Immediately
		E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

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SURVEILLANCE REQUIREMENTS

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NOTE

CTS 4.2 1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.

DOC L.3 2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.

CTS		SURVEILLANCE	FREQUENCY
Table 4.2.1 Functions 5 and 6, DOC M.1	SR 3.3.2.1.1	Perform CHANNEL FUNCTIONAL TEST.	[92] days
4.3.B.3.(a)(ii), 4.3.B.3.(a)(iii), 4.3.B.3.(a)(iv)	SR 3.3.2.1.2	NOTE Not required to be performed until 1 hour after any control rod is withdrawn at ≤ [10]% RTP in MODE 2. Perform CHANNEL FUNCTIONAL TEST.	[92] days
4.3.B.3.(a)(ii), 4.3.B.3.(a)(iii), 4.3.B.3.(a)(iv)	SR 3.3.2.1.3	NOTE Not required to be performed until 1 hour after THERMAL POWER is ≤ [10]% RTP in MODE 1. Perform CHANNEL FUNCTIONAL TEST.	[92] days

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SURVEILLANCE REQUIREMENTS (continued)

CTS	SURVEILLANCE		FREQUENCY	
Table 4.2.1 Functions 5 and 6	SR 3.3.2.1.~~4~~ 5	-----NOTE----- [Neutron detectors are excluded.] ----- Verify the RBM: a. Low Power Range - Upscale Function is not bypassed when THERMAL POWER is ≥~~29~~% ≥ 30 and ≤~~64~~% < 65 RTP~~.~~ ; b. Intermediate Power Range - Upscale Function is not bypassed when THERMAL POWER is >~~64~~% ≥ 65 and ≤~~84~~% < 85 RTP~~.~~ ; and c. High Power Range - Upscale Function is not bypassed when THERMAL POWER is >~~84~~% ≥ 85 RTP.	~~[18] months~~ 92 days	2 2 7 7 3
DOC M.3	SR 3.3.2.1.~~5~~ 6	Verify the RWM is not bypassed when THERMAL POWER is ≤ [10]% RTP.	[~~18~~] 24 months	2 1
DOC M.2	SR 3.3.2.1.~~6~~ 7	-----NOTE----- Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. ----- Perform CHANNEL FUNCTIONAL TEST.	[~~18~~] 24 months	2 1
Table 4.2.1 Functions 5 and 6	SR 3.3.2.1.~~7~~ 4	-----NOTE----- Neutron detectors are excluded. ----- Perform CHANNEL CALIBRATION.	~~[18] months~~ 92 days 1	2
4.3.B.3.(a)(l)	SR 3.3.2.1.8	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM	

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Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

CTS	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	1 REQUIRED CHANNELS	2 SURVEILLANCE REQUIREMENTS	1 ALLOWABLE VALUE
	1. Rod Block Monitor				As specified in COLR
3.2.C.2.a, Table 3.2.3 Function 4.a including Note (8), Table 4.2.1 Function 5	a. Low Power Range - Upscale	(a)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.4 5 SR 3.3.2.1.7	≤ [115.5/125] divisions of full scale
	b. Intermediate Power Range - Upscale	(b)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.4 5 SR 3.3.2.1.7	≤ [109.7/125] divisions of full scale
	c. High Power Range - Upscale	(c),(d)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.4 5 SR 3.3.2.1.7	≤ [105.9/125] divisions of full scale
DOC M.1	d. Inop	(d),(e)	[2]	SR 3.3.2.1.1	NA
3.2.C.2.a, Table 3.2.3 Function 4.b, Table 4.2.1 Function 6	e. Downscale	(d),(e)	[2]	SR 3.3.2.1.1 4 SR 3.3.2.1.7	≥ [93/125] 93.6 divisions of full scale
	~~f. Bypass Time Delay~~	~~(d),(e)~~	~~[2]~~	~~SR 3.3.2.1.1~~ ~~SR 3.3.2.1.7~~	~~≤ [2.0] seconds~~ 6
3.3.B.3.b, 4.3.B.3.a,	2. Rod Worth Minimizer	1(f), 2(f)	[1]	SR 3.3.2.1.2 SR 3.3.2.1.3 6 SR 3.3.2.1.5 SR 3.3.2.1.8	NA
1.0.Y, DOC M.2	3. Reactor Mode Switch - Shutdown Position	(g)	[2]	SR 3.3.2.1.6 7	NA

3.2.C.2.b, Table 3.2.3 Note (8)

(a) THERMAL POWER ≥ ~~[29]~~ 30 % and ~~≤ [64]~~ < 65 % RTP and MCPR ~~< 1.70~~ is below the limit specified in COLR 3

(b) THERMAL POWER ~~> [64]~~ ≥ 65 % and ~~≤ [84]~~ < 85 % RTP and MCPR ~~< 1.70~~.

(c) THERMAL POWER ~~> [84]~~ ≥ 85 % and < 90% RTP and MCPR ~~< 1.70~~.

(d) THERMAL POWER ≥ 90% RTP and MCPR ~~< 1.40~~.

3.2.C.2.a (e) THERMAL POWER ≥ ~~[64]~~ 30 % and < 90% RTP and MCPR ~~< 1.70~~.

3.3.B.3.b (f) With THERMAL POWER ≤ [10]% RTP.

1.0.Y, DOC M.2 (g) Reactor mode switch in the shutdown position.

1

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.2.1, CONTROL ROD BLOCK INSTRUMENTATION

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Monticello Rod Block Monitor Functions are required to be calibrated by the CTS at a Frequency of 92 days. Therefore, the bracketed Frequency of ISTS SR 3.3.1.2.7 has been revised and ISTS SR 3.3.1.2.7 has been renumbered as ITS SR 3.3.2.1.4 consistent with the ITS format. Subsequent SRs have been renumbered to reflect this change.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

4. ISTS 3.3.2.1 Required Action C.2.1.2 has been modified to be consistent with the Bases. This is also consistent with changes approved by the NRC for the Quad Cities 1 and 2, Dresden 2 and 3, and LaSalle 1 and 2 ITS conversions, and with proposed TSTF-464.

5. Typographical error corrected.

6. ISTS Table 3.3.2.1-1 requires the RBM Bypass Time Delay Function (Function 1.f) to be OPERABLE, and provides the applicable Surveillances to verify OPERABILITY. After the RBM upscale trip exceeds the trip setpoint, the generated control rod block is allowed to be delayed for a short time by the RBM Bypass Time Delay Function, prior to sending the control rod block signal to the associated Reactor Manual Control System rod block circuit. The safety analysis does not require a time delay, but only assumes the signal is delayed for up to 2.0 seconds. The RBM includes an electronic "dip" switch that bypasses the RBM Bypass Time Delay Function. When the switch is placed in the bypass position, the RBM Bypass Time Delay Function is effectively removed from the RBM circuitry; i.e., the time delay is set to zero seconds. Monticello performed a modification to the RBM System a few years ago (as part of the APRM, RBM, and Technical Specifications modification), and at that time set the "dip" switch to the bypass position as a permanent feature of the modification. Thus, the RBM control rod block signal is not currently being delayed. Therefore, the allowance to have a RBM Bypass Time Delay (set at up to 2.0 seconds) is not needed and has been deleted.

7. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch - Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. It is assumed to function to block further control rod withdrawal to preclude a MCPR Safety Limit (SL) violation. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. A signal from one average power range monitor (APRM) channel assigned to each Reactor Protection System (RPS) trip system supplies a reference signal for the RBM channel in the same trip system. This reference signal is used to determine which RBM range setpoint (low, intermediate, or high) is enabled. If the APRM is indicating less than the low power range setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 1).

INSERT 1 1

The purpose of the RWM is to control rod patterns during startup, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences

1 **INSERT 1**

Furthermore, the Bypass Time Delay, which bypasses the RBM upscale trips for a short period of time, is not utilized (it is permanently disabled). Thus, if it is not disabled, the associated RBM channel is inoperable. In addition, to preclude rod movement with an inoperable RBM, a downscale trip and an inoperable trip are provided. A RBM channel is considered inoperable if less than half the total number of inputs are available.

BASES

BACKGROUND (continued)

are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence based position indication for each control rod. The RWM also uses feedwater flow and steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 2). The RWM is a single channel system that provides input into both RMCS rod block circuits. 1

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This Function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 3. A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

The RBM Function satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range, to ensure that no single instrument failure can preclude a rod block from this Function. The actual setpoints are calibrated consistent with applicable setpoint methodology.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. ~~The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.~~ INSERT 2 1

≥ 30 The RBM is assumed to mitigate the consequences of an RWE event when operating ~~≥ 29~~% RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating < 90% RTP, 5 analyses ~~(Ref. 3)~~ have shown that with an initial MCPR ≥ 1.7~~0~~, no RWE event will result in exceeding the MCPR SL. Also, the analyses 4 demonstrate that when operating at ≥ 90% RTP with MCPR ≥ 1.4~~0~~, no RWE event will result in exceeding the MCPR SL ~~(Ref. 3)~~. Therefore, under these conditions, the RBM is also not required to be OPERABLE. 1

2. Rod Worth Minimizer

The RWM enforces the banked position withdrawal sequence (BPWS) to ensure that the initial conditions of the CRDA analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References 4, 5, 6, and 7. The BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

The RWM Function satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

1

INSERT 2

The Allowable Values and nominal trip setpoints (NTSP) are derived, using the General Electric setpoint methodology guidance, as specified in the Monticello setpoint methodology. The Allowable Values are derived from the analytic limits. The difference between the analytic limit and the Allowable Value allows for channel instrument accuracy, calibration accuracy, process measurement accuracy, and primary element accuracy. The margin between the Allowable Value and the NTSP allows for instrument drift that might occur during the established surveillance period. Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift. These two verifications are statistical evaluations to provide additional assurance of the acceptability of the NTSP and may require changes to the NTSP. Use of these methods and verifications provides the assurance that if the setpoint is found conservative to the Allowable Value during surveillance testing, the instrumentation would have provided the required trip function by the time the process reached the analytic limit for the applicable events.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Since the RWM is a hardwired system designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. 7). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the BPWS. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed. 1

Compliance with the BPWS, and therefore OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is ~~<~~ ≤ 10% RTP. When THERMAL POWER is > 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Refs. 5 and 7). In MODES 3 and 4, all control rods are required to be inserted into the core; therefore, a CRDA cannot occur. In MODE 5, since only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will be subcritical. 2

3. Reactor Mode Switch - Shutdown Position

During MODES 3 and 4, and during MODE 5 when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch - Shutdown Position control rod withdrawal block ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby preserving the assumptions of the safety analysis.

The Reactor Mode Switch - Shutdown Position Function satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Two channels are required to be OPERABLE to ensure that no single channel failure will preclude a rod block when required. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on reactor mode switch position.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued) when the reactor mode switch is in the shutdown position

s and and MODE

During shutdown conditions (MODE 3, 4, ~~or~~ 5), no positive reactivity insertion events are analyzed because assumptions are that control rod withdrawal blocks are provided to prevent criticality. Therefore, when the reactor mode switch is in the shutdown position, the control rod withdrawal block is required to be OPERABLE. During MODE 5 with the reactor mode switch in the refueling position, the refuel position one-rod-out interlock (LCO 3.9.2) provides the required control rod withdrawal blocks. 7 4

, "Refuel Position One-Rod-Out Interlock"

ACTIONS

~~REVIEWER'S NOTE~~
~~Certain LCO Completion Times are based on approved topical reports. In order for the licensee to use the times, the licensee must justify the Completion Times as required by the staff Safety Evaluation Report (SER) for the topical report.~~ 5

A.1

With one RBM channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod block function; however, overall reliability is reduced because a single failure in the remaining OPERABLE channel can result in no control rod block capability for the RBM. For this reason, Required Action A.1 requires restoration of the inoperable channel to OPERABLE status. The Completion Time of 24 hours is based on the low probability of an event occurring coincident with a failure in the remaining OPERABLE channel.

B.1

If Required Action A.1 is not met and the associated Completion Time has expired, the inoperable channel must be placed in trip within 1 hour. If both RBM channels are inoperable, the RBM is not capable of performing its intended function; thus, one channel must also be placed in trip. This initiates a control rod withdrawal block, thereby ensuring that the RBM function is met.

The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping of inoperable channels.

BASES

ACTIONS (continued)

C.1, C.2.1.1, C.2.1.2, and C.2.2

With the RWM inoperable during a reactor startup, the operator is still capable of enforcing the prescribed control rod sequence. However, the overall reliability is reduced because a single operator error can result in violating the control rod sequence. Therefore, control rod movement must be immediately suspended except by scram. Alternatively, startup may continue if at least 12 control rods have already been withdrawn, or a reactor startup with an inoperable RWM was not performed in the last 12 months. Required Actions C.2.1.1 and C.2.1.2 require verification of these conditions by review of plant logs and control room indications. Once Required Action C.2.1.1 or C.2.1.2 is satisfactorily completed, control rod withdrawal may proceed in accordance with the restrictions imposed by Required Action C.2.2. Required Action C.2.2 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified 9
member of the technical staff. 1
(engineer)

The RWM may be bypassed under these conditions to allow continued operations. In addition, Required Actions of LCO 3.1.3 and LCO 3.1.6 may require bypassing the RWM, during which time the RWM must be considered inoperable with Condition C entered and its Required Actions taken.

D.1

With the RWM inoperable during a reactor shutdown, the operator is still capable of enforcing the prescribed control rod sequence. Required Action D.1 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor 9
Operator) or other qualified member of the technical staff. The RWM may be bypassed under these conditions to allow the reactor shutdown to continue. (engineer) 1

BASES

ACTIONS (continued)

E.1 and E.2

With one Reactor Mode Switch - Shutdown Position control rod withdrawal block channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod withdrawal block function. However, since the Required Actions are consistent with the normal action of an OPERABLE Reactor Mode Switch - Shutdown Position Function (i.e., maintaining all control rods inserted), there is no distinction between having one or two channels inoperable.

, "SHUTDOWN MARGIN (SDM)

In both cases (one or both channels inoperable), suspending all control rod withdrawal and initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies will ensure that the core is subcritical with adequate SDM ensured by LCO 3.1.1. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are therefore not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted. 6

SURVEILLANCE REQUIREMENTS

~~REVIEWER'S NOTE~~
~~Certain Frequencies are based on approved topical reports. In order for a licensee to use these Frequencies, the licensee must justify the Frequencies as required by the staff SER for the topical report.~~ 5

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

second

The Surveillances are modified by a Note to indicate that when an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 9) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary. 6

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control Multiplexing System input. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. 1

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of 92 days is based on reliability analyses (Ref. 8).

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The CHANNEL FUNCTIONAL TEST for the RWM is performed by : a) attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs; b) verifying proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group; and c) performing a RWM computer on-line diagnostic test. ~~As noted in the SRs,~~ SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2, and ~~As noted,~~ SR 3.3.2.1.3 is not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. This allows entry into MODE 2 for SR 3.3.2.1.2, and entry into MODE 1 when THERMAL POWER is ≤ 10% RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 8). 1 7 6

BASES

SURVEILLANCE REQUIREMENTS (continued)

INSERT from page B 3.3.2.1-10

SR 3.3.2.1.4 5 (3)

The RBM setpoints are automatically varied as a function of power. Three Allowable Values ~~are specified~~ in Table 3.3.2.1-1, each within a specific power range. The power at which the control rod block Allowable Values automatically change are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These ~~power Allowable Values~~ must be verified periodically to be less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.6. The ~~18 month~~ Frequency is based on the actual trip setpoint methodology utilized for these channels. (3) (1) (3)

required

, are specified in the COLR

bypass setpoints

92 day

SR 3.3.2.1.5 6 (3)

The RWM is automatically bypassed when power is above a specified value. The power level is determined from ~~feedwater flow and~~ steam flow signals. The automatic bypass setpoint must be verified periodically to be ≤ [10]% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on ~~the trip setpoint methodology utilized for the low power setpoint channel~~. (1) (2) (8) (6) (1)

>

24 month

INSERT 3

SR 3.3.2.1.6 7 (3)

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch - Shutdown Position Function to ensure that the entire channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an

1 **INSERT 3**

engineering judgment considering the reliability of the components, and that indication of whether or not the RWM is bypassed is provided in the control room.

BASES

SURVEILLANCE REQUIREMENTS (continued)

acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch - Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the 24 [18] month Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. 3

24 The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 [18] month Frequency. 3 3

move to page B 3.3.2.1-9 as indicated

SR 3.3.2.1.~~[7]~~ 4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. 3

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.6. 3

The Frequency is based upon the assumption of a ~~n 18 month~~ 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. 3

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.8

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

REFERENCES

U

1. FSAR, Section [7.6.2.2.5]. 7.3.5.3 8

U

2. FSAR, Section [7.6.8.2.6]. 7.8.2 8

92

3. NEDC-30474-P, "Average Power Range Monitor, Rod Block Monitor, and Technical Specification Improvements (ARTS) Program for Edwin I. ~~Hatch Nuclear~~ Plants," ~~December 1983~~. April 1984

Monticello Nuclear Generating Plant

4. NEDE-24011-P-A-9-US, "General Electrical Standard Application for Reload Fuel," Supplement for ~~United States~~, Section S 2.2.3.1, ~~September 1988~~. (revision specified in Specification 5.6.3)

5. "Modifications to the Requirements ~~for Control~~ Rod Drop Accident Mitigating Systems," ~~BWR Owners'~~ Group, July 1986.

6. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

7. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.

8. NEDC-30851-P-A, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

-A

9. GENE-770-06-1, "~~Addendum to~~ Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," ~~February 1991~~. December 1992

Letter from T.A. Pickens (BWROG) to G.C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1986.

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JUSTIFICATION FOR DEVIATIONS
ITS 3.3.2.1 BASES, CONTROL ROD BLOCK INSTRUMENTATION

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. Typographical/Grammatical error corrected.

3. Changes have been made to reflect those changes made to the Specification.

4. The Title's of the LCO's have been included the first time it appears in the LCO Bases to be consistent with other places in the Bases.

5. This Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

6. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

7. Changes have been made to more closely reflect the Specification requirements.

8. The brackets have been removed and the proper plant specific information/value has been provided.

9. Typographical error corrected. The terms in 10 CFR 55.4 and 10 CFR 50.54(m) are "Senior Operator" and "Operator," not "Senior Reactor Operator" and "Reactor Operator."

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.2.1, CONTROL ROD BLOCK INSTRUMENTATION

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGE L.5

Nuclear Management Company, LLC (NMC) is converting the Monticello Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) as outlined in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4." The proposed change involves making the Current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1433.

CTS 3.2.C.2.b states that the RBM bypass time delay must be less than or equal to 2.0 seconds. ITS 3.3.2.1 does not require the RBM bypass time delay to be OPERABLE. This changes the CTS by deleting the RBM bypass time delay requirements.

The purpose of CTS 3.2.C.2.b is to ensure that the RBM bypass time delay is within the assumed limit. After the RBM upscale trip exceeds the trip setpoint, the generated control rod block is allowed to be delayed for a short time (up to 2 seconds as allowed by CTS 3.2.C.2.b) by the RBM bypass time delay, prior to sending the control rod block signal to the associated Reactor Manual Control System rod block circuit. The safety analysis does not require a time delay, but only assumes the signal is delayed for up to 2.0 seconds. The RBM includes an electronic "dip" switch that bypasses the RBM bypass time delay feature. When the switch is placed in the bypass position, the RBM bypass time delay is effectively removed from the RBM circuitry; i.e., the time delay is set to zero seconds. Monticello performed a modification to the RBM System a few years ago (as part of the APRM, RBM, and Technical Specifications modification), and at that time set the "dip" switch to the bypass position as a permanent feature of the modification. Thus, the RBM control rod block signal is not currently being delayed. Therefore, the CTS allowance to have a RBM bypass time delay (set at up to 2.0 seconds) is not needed. This change is less restrictive because the LCO requirement for a RBM bypass time delay feature is being removed from the CTS.

NMC has evaluated whether or not a significant hazards consideration is involved with these proposed Technical Specification changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed change deletes the allowance for a RBM bypass time delay. The safety analysis allows the RBM rod block signal to be delayed for up to 2.0 seconds. With the RBM bypass time delay permanently disabled, the RBM rod block signal will not be delayed. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.2.1, CONTROL ROD BLOCK INSTRUMENTATION

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change deletes the allowance for a RBM bypass time delay. The safety analysis allows the RBM rod block signal to be delayed for up to 2.0 seconds. With the RBM bypass time delay permanently disabled, the RBM rod block signal will not be delayed. The position of the "dip" switch, which disabled the RBM bypass time delay, was verified during the modification process. Therefore, this change (to remove the RBM bypass time delay allowance form the Technical Specifications) will not physically alter the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change deletes the allowance for a RBM bypass time delay. The safety analysis allows the RBM rod block signal to be delayed for up to 2.0 seconds. With the RBM bypass time delay permanently disabled, the RBM rod block signal will not be delayed. The safety analysis does not require a time delay, but only assumes the signal is delayed for up to 2.0 seconds. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

ATTACHMENT 4

ITS 3.3.2.2, Feedwater Pump and Main Turbine High Water Level Instrumentation

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

Add proposed ITS 3.3.2.2

M.1

DISCUSSION OF CHANGES
ITS 3.3.2.2, FEEDWATER PUMP AND MAIN TURBINE HIGH WATER LEVEL TRIP INSTRUMENTATION

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

M.1 The CTS does not have any specific requirements for the Feedwater Pump and Main Turbine High Water Level Trip Instrumentation. ITS LCO 3.3.2.2 requires four channels of Feedwater Pump and Main Turbine High Water Level Trip Instrumentation to be OPERABLE. Appropriate ACTIONS and Surveillance Requirements are also provided. This changes the CTS by incorporating the requirements of ISTS 3.3.2.2.

The Feedwater Pump and Main Turbine High Water Level Trip Instrumentation is necessary to mitigate the Feedwater Controller Failure Maximum Demand event (a design basis transient). The Feedwater Pump and Main Turbine High Water Level Trip Instrumentation trips the feedwater pumps, thereby limiting any further feedwater addition and increases in reactor water level, and trips the main turbine, which closes the turbine stop valves, thereby preventing turbine damage due to water entering the turbine. In addition, the turbine stop valve closure initiates a reactor scram to mitigate the reduction in MCPR. The requirement to maintain four feedwater pump and main turbine high water level trip channels OPERABLE ensures that no single instrument failure will prevent the feedwater pumps and main turbine high water level trip on a valid high level signal. This change is acceptable because the Feedwater Pump and Main Turbine High Water Level Trip Instrumentation detects a potential failure of the Feedwater level Control System to prevent further reactor vessel water level increases, protects the main turbine from water damage, and indirectly initiates a reactor scram to mitigate the reduction in MCPR. The ITS 3.3.2.2 ACTIONS ensure sufficient high level trip channels are OPERABLE or trip systems are configured to assure a trip of the feedwater pumps and main turbine when a feedwater pump and main turbine high level trip channel(s) are inoperable. In addition, specific Surveillance requirements are now specified. This change is more restrictive because it adds new requirements to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.3 INSTRUMENTATION

Pump

3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation ①

DOC M.1 LCO 3.3.2.2 Four ~~[Three]~~ channels of Pump feedwater and main turbine high water level trip instrumentation shall be OPERABLE. ② ①

APPLICABILITY: THERMAL POWER ≥ [25]% RTP. ②

ACTIONS

-----NOTE-----

DOC M.1 Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOC M.1 A. One or more feedwater pump and main turbine high water level trip channels inoperable.	A.1 Place channel in trip.	7 days ③
DOC M.1 B. ~~Two or more~~ feedwater pump and main turbine high water level trip ~~channels inoperable~~ capability not maintained.	B.1 Restore feedwater pump and main turbine high water level trip capability.	2 hours ③
DOC M.1 C. Required Action and associated Completion Time not met.	C.1 -----NOTE----- Only applicable if inoperable channel is the result of inoperable feedwater pump breaker ~~[valve]~~ or main turbine stop valve. ----- Remove affected feedwater pump(s) and main turbine valve(s) from service. OR	4 hours ②

Pump

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	C.2 Reduce THERMAL POWER to < [25]% RTP.	4 hours 2

SURVEILLANCE REQUIREMENTS

NOTE

DOC M.1 When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided feedwater and main turbine high water level trip capability is maintained. 1

pump

	SURVEILLANCE		FREQUENCY
DOC M.1	SR 3.3.2.2.1	[Perform CHANNEL CHECK.	12 [24 hours] 2
DOC M.1	SR 3.3.2.2.2	Perform CHANNEL FUNCTIONAL TEST.	184 [92] days 4
DOC M.1	SR 3.3.2.2.3 4	Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ [58.0] 49 inches.	[18] 24 months 4 2
DOC M.1	SR 3.3.2.2.4 5	Perform LOGIC SYSTEM FUNCTIONAL TEST including [valve] and breaker actuation.	[18] 24 months 4 2
DOC M.1	SR 3.3.2.2.3	Calibrate the trip units.	184 days 4

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.2.2, FEEDWATER PUMP AND MAIN TURBINE HIGH WATER LEVEL TRIP INSTRUMENTATION

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. ISTS 3.3.2.2 ACTIONS A and B are written for a three channel design with a two-out-of-three logic design. For this three channel design, when two of the three channels are inoperable, a loss of function has occurred. The Monticello Feedwater Pump and Main Turbine High Water Level Trip Instrumentation includes four channels, with a one-out-of-two-taken-twice logic design. Thus, the Monticello design is such that with two channels inoperable, a loss of function may not have occurred. Therefore, ISTS 3.3.2.2 Condition A has been revised to be applicable to one or more inoperable channels, and ISTS 3.3.2.2 Condition B has been revised to be applicable to when a loss of function has occurred (i.e., trip capability not maintained). This change is consistent with the intent of the ISTS, which requires the 2 hour Completion Time of ACTION B to be applicable when a loss of function has occurred.

4. ITS SR 3.3.2.2.3 has been added to include calibration of the trip units every 184 days, consistent with the specific plant instrumentation design and current practice at Monticello. Subsequent Surveillance Requirements have been renumbered to reflect this change. In addition, the Frequency for ISTS SR 3.3.2.2.2 has been changed to 184 days, consistent with current practice.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION

Pump

B 3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation (1)

BASES

BACKGROUND

Pump

The feedwater and main turbine high water level trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow. (2)

With excessive feedwater flow, the water level in the reactor vessel rises toward the high water level ~~, Level 8~~ reference point, causing the trip of the two feedwater pump ~~turbines~~ s and the main turbine.

Reactor Vessel Water Level - High ~~, Level 8~~ signals are provided by level sensors that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level in the reactor vessel (variable leg). Four ~~Three~~ channels of Reactor Vessel Water Level - High ~~, Level 8~~ instrumentation are provided as input to a one-out-of-two-taken-twice ~~two-out-of-three~~ initiation logic that trips the two feedwater pump s ~~turbines~~ and the main turbine. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre- established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a ~~main~~ feedwater pump and main turbine trip signal to the trip logic. (3)

A trip of the feedwater pump s ~~turbines~~ limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the stop valves protects the turbine from damage due to water entering the turbine.

APPLICABLE SAFETY ANALYSES

Pump

The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The high ~~Level 8~~ trip indirectly initiates a reactor scram from the main turbine trip (above 45 ~~30~~% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR. (2) (3)

Pump

Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). (2)

LCO

The LCO requires four ~~three~~ channels of the Reactor Vessel Water Level - High ~~, Level 8~~ instrumentation to be OPERABLE to ensure that no single instrument failure will prevent the feedwater pump s ~~turbines~~ and main turbine trip on a valid high ~~Level 8~~ signal. ~~Two of the three channels are~~ (3) (2)

BASES

LCO (continued)

needed to ~~provide trip~~ signals in order for ~~the feedwater and~~ main turbine ~~trips to occur.~~ Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.3. The Allowable Value is set to ensure that the thermal limits are not exceeded during the event. The actual setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. 2 4 1

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. ~~The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.~~ INSERT 1 3

APPLICABILITY

Pump

The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," ~~and~~ LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," sufficient margin to these limits exists below 25% RTP; therefore, these requirements are only necessary when operating at or above this power level. 2 2 3

and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR),"

3

INSERT 1

The Allowable Values and nominal trip setpoints (NTSP) are derived, using the General Electric setpoint methodology guidance, as specified in the Monticello setpoint methodology. The Allowable Values are derived from the analytic limits. The difference between the analytic limit and the Allowable Value allows for channel instrument accuracy, calibration accuracy, process measurement accuracy, and primary element accuracy. The margin between the Allowable Value and the NTSP allows for instrument drift that might occur during the established surveillance period. Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift. These two verifications are statistical evaluations to provide additional assurance of the acceptability of the NTSP and may require changes to the NTSP. Use of these methods and verifications provides the assurance that if the setpoint is found conservative to the Allowable Value during surveillance testing, the instrumentation would have provided the required trip function by the time the process reached the analytic limit for the applicable events.

 1

BASES

ACTIONS

A Note has been provided to modify the ACTIONS related to feedwater Pump and main turbine high water level trip instrumentation channels. 2
Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions , subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable feedwater and main turbine high water level trip instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable feedwater Pump and main turbine high water level trip instrumentation channel. 7 2

A.1

With one or more channel s inoperable and trip capability maintained, the remaining ~~two~~ OPERABLE channels can provide the required trip signal. However, overall instrumentation reliability is reduced because a single failure in one of the remaining channels concurrent with feedwater controller failure, maximum demand event, may result in the instrumentation not being able to perform its intended function. Therefore, continued operation is only allowed for a limited time ~~with one channel inoperable~~. If the inoperable channel (s) cannot be restored to OPERABLE status within the Completion Time, the channel (s) must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in a feedwater pump or main turbine trip), Condition C must be entered and its Required Action taken. 3

The Completion Time of 7 days is based on the low probability of the event occurring coincident with a single failure in a remaining OPERABLE channel.

1

BASES

ACTIONS (continued)

B.1

Pump

With ~~two or more channels inoperable~~, the feedwater and main turbine high water level trip instrumentation cannot perform its design function (feedwater and main turbine high water level trip capability [is] not maintained). Therefore, continued operation is only permitted for a 2 hour period, during which feedwater and main turbine high water level trip capability must be restored. The trip capability is considered maintained when sufficient channels are OPERABLE or in trip such that the feedwater and main turbine high water level trip logic will generate a trip signal on a valid signal. This requires ~~two channels to each be OPERABLE or in trip.~~ If the required channels cannot be restored to OPERABLE status or placed in trip, Condition C must be entered and its Required Action taken. 2 3

the

pump

pump

Trip capability is lost if two parallel contacts (channels) in the same trip system are inoperable and not tripped.

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of feedwater and main turbine high water level trip instrumentation occurring during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation. 2

Pump

C.1 and C.2

With the required channels not restored to OPERABLE status or placed in trip, THERMAL POWER must be reduced to < 25% RTP within 4 hours. Alternatively, the affected feedwater pump(s) and affected main turbine valve(s) may be removed from service since this performs the intended function of the instrumentation. As discussed in the Applicability section of the Bases, operation below [25%] RTP results in sufficient margin to the required limits, and the feedwater and main turbine high water level trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < 25% RTP from full power conditions in an orderly manner and without challenging plant systems. 2

Pump

Required Action C.1 is modified by a Note which states that the Required Action is only applicable if the inoperable channel is the result of an inoperable feedwater pump ~~[valve]~~ or main turbine stop valve. The Note clarifies the situations under which the associated Required Action would be the appropriate Required Action. 6

breaker

 1

BASES

SURVEILLANCE REQUIREMENTS

~~REVIEWER'S NOTE~~
~~Certain Frequencies are based on approved topical reports. In order for a licensee to use these Frequencies the licensee must justify the Frequencies as required by the staff Safety Evaluation Report (SER) for the topical report.~~ 4

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater pump and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 2) assumption that 6 hours is the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the feedwater pumps ~~turbines~~ and main turbine will trip when necessary. 3 3

SR 3.3.2.2.1

Performance of the CHANNEL CHECK once every 12 ~~24~~ hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels, or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. 1

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limits.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel status during normal operational use of the displays associated with the channels required by the LCO.

 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. 2

184

The Frequency of ~~92~~ days is based on ~~reliability analysis (Ref. 2)~~. 1

INSERT 2 5

engineering judgment and the reliability of these components

4

SR 3.3.2.2.~~3~~ 5

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

24

The Frequency is based upon the assumption of a~~n~~ ~~18~~ month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. 1

5

SR 3.3.2.2.~~4~~ 5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. ~~The system functional test of the feedwater and main turbine valves is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a valve is incapable of operating, the associated instrumentation would also be inoperable.~~ The ~~18~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the ~~18~~ month Frequency. 1 3 1 1

pump breakers

stop

24

5

INSERT 2

SR 3.3.2.2.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in SR 3.3.2.2.4. If the trip setting is discovered to be less conservative than the setting accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 184 days is based on engineering judgment and the reliability of these components.

BASES

REFERENCES (3) (6)

U

1. ~~F~~SAR, Section ~~[15.1]~~ 14.4.4.

-A

2. GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications," ~~February 1991~~ December 1992. (3)

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.2.2 BASES, FEEDWATER PUMP AND MAIN TURBINE HIGH WATER LEVEL TRIP INSTRUMENTATION

1. Changes have been made to reflect those changes made to the Specification.

2. Editorial changes made to be consistent with similar statements in other places in the Bases.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

4. This Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

5. ITS SR 3.3.2.2.3 has been added to include calibration of the trip units every 184 days, consistent with the specific plant instrumentation design and current practice at Monticello. Subsequent Surveillance Requirements have been renumbered to reflect this change.

6. The brackets have been removed and the proper plant specific information/value has been provided.

7. Typographical error corrected.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.2.2, FEEDWATER PUMP AND MAIN TURBINE HIGH WATER LEVEL TRIP INSTRUMENTATION

There are no specific NSHC discussions for this Specification.

ATTACHMENT 5

ITS 3.3.3.1, Post Accident Monitoring (PAM) Instrumentation

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3.14 ~~ACCIDENT MONITORING INSTRUMENTATION~~	4.14 ~~ACCIDENT MONITORING INSTRUMENTATION~~
~~Applicability:~~	~~Applicability:~~
~~Applies to plant instrumentation which does not perform a protective function, but which provides information to monitor and assess important parameters during and following an accident.~~	~~Applies to the surveillance requirements for accident monitoring instrumentation.~~
~~Objective:~~	~~Objective:~~
~~To ensure that sufficient information is available to operators to determine the effects of and determine the course of an accident to the extent required to carry out required manual actions.~~	~~To specify the type and frequency of surveillance to be applied to accident monitoring instrumentation.~~
Specification:	Specification:
Applicability — Whenever irradiated fuel is in the reactor vessel and reactor coolant water temperature is greater than 212°F, the limiting	M.1 SRs Note
LCO 3.3.3.1 — conditions for operation for accident monitoring instrumentation given in Table 3.14.1 shall be satisfied.	L.1 The accident monitoring instrumentation shall be functionally tested and calibrated in accordance with Table 4.14.1.
	Add proposed Surveillance Requirements Note 2 — L.2

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A.1

ITS

Table 3.3.3.1-1

Table 3.14.1

Instrumentation for Accident Monitoring

ITS	Function	Total No. of Instrument Channels	Minimum No. of Operable Channels	Required Conditions*
2	Reactor Vessel Fuel Zone Water Level	2	1	A, B
	~~Safety/Relief Valve Position (One Channel Pressure Switch and One Channel Thermocouple Position Indication per Valve)~~	~~2~~	~~1~~	~~A, C~~
4	Drywell Wide Range Pressure	2	1	A, B
3	Suppression Pool Wide Range Level	2	1	A, B
7	Suppression Pool Temperature	2	1	A, D
5	Drywell High Range Radiation	2	1	A, D
	~~Offgas Stack Wide Range Radiation~~	~~2~~	~~1~~	~~A, D~~
	~~Reactor Bldg Vent Wide Range Radiation~~	~~2~~	~~1~~	~~A, D~~

R.1

R.1

* Required Conditions

Add proposed ACTIONS Note — A.2

30 — L.3

ACTION A — A. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the total number of channels, either restore the inoperable channels to operable status within seven days, or prepare and submit a special

ACTION B — report to the Commission pursuant to Technical Specification 6.7.D within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status.

See ITS 5.6

Add proposed Functions 1 and 6 — M.2

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A.1

ITS

Table 3.3.3.1-1

Table 3.14.1 (Continued)

Instrumentation for Accident Monitoring

* Required Conditions (continued)

7 days L.4

ACTION C — B. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, the minimum number of channels shall be restored to operable status within 48 ~~hours~~

ACTIONS D and E — or be in at least Hot Shutdown within the next 12 hours ~~and Cold Shutdown within the following 24 hours~~ L.1

~~C. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, the torus temperature shall be monitored once per 12 hours (+25%) to observe any unexplained temperature increase which might be indicative of an open SRV; the minimum number of channels shall be restored to operable status within 30 days or be in at least Hot Shutdown within the next 12 hours and Cold Shutdown within the following 24 hours.~~ R.1

restore in 7 days L.5

ACTION C — D. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, ~~initiate the preplanned alternate method of monitoring the appropriate parameters~~ in LA.1

ACTIONS D and F — addition to submitting the report required in (A) above.

A.1

ITS

Table 3.3.3.1-1

Table 4.14.1

Minimum Test and Calibration Frequency for Accident Monitoring Instrumentation

A.4 24 months

SR 3.3.3.1.2 SR 3.3.3.1.1 A.3

ITS	Instrument Channel	Test (Note 1)	Calibration (Note 1)	Sensor Check (Note 1)	
2	Reactor Vessel Fuel Zone Water Level Monitor	-	Once/Operating Cycle	Once/month (Note 3)	A.5
	~~Safety/Relief Valve Position (Pressure Switches)~~	-	~~Once/Operating Cycle~~	~~Once/month (Notes 2 & 4)~~	R.1
	~~Safety/Relief Valve Position (Thermocouples)~~	-	~~Once/Operating Cycle~~	~~Once/month (Note 4)~~	
4	Drywell Wide Range Pressure Monitors	-	Once/Operating Cycle	Once/month	24 months A.4
3	Suppression Pool Wide Range Level Monitors	-	Once/Operating Cycle	Once/month	
7	Suppression Pool Temperature	-	Once/Operating Cycle	Once/month	
5	Drywell High Range Radiation Monitors	-	Once/Operating Cycle	Once/month	
	~~Offgas Stack Wide Range Radiation Monitors~~	-	~~Once/Operating Cycle~~	~~Once/month~~	R.1
	~~Reactor Bldg Wide Range Radiation Monitors~~	-	~~Once/Operating Cycle~~	~~Once/month~~	

Notes:

~~(1) Functional tests, calibrations, and sensor checks are not required when the instruments are not required to be operable. If tests are missed, they shall be performed prior to returning the instruments to an operable status.~~ A.3

~~(2) Once/month sensor check will consist of verifying that the pressure switches are not tripped.~~ R.1

~~(3) Once/month sensor check will consist of verifying that the fuel zone level indicates off scale high.~~ A.5

~~(4) Following every Safety/Relief Valve actuation it will be verified that recorder traces or computer logs indicate sensor responses.~~ R.1

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DISCUSSION OF CHANGES
ITS 3.3.3.1, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS Table 3.14.1 Required Actions A, B, and D specify the compensatory actions to take when PAM Instrumentation is inoperable. ITS 3.3.3.1 ACTIONS provide the compensatory actions for inoperable PAM Instrumentation. The ITS 3.3.3.1 ACTIONS include a Note that allows separate Condition entry for each Function. This modifies the CTS by providing a specific allowance to enter the Action for each inoperable PAM instrumentation Function.

This change is acceptable because it clearly states the current requirement. The CTS considers each PAM instrumentation Function to be separate and independent from the others. This change is administrative because it does not result in technical changes to the CTS.

A.3 CTS Table 4.14.1 Note (1) states that functional tests, calibrations, and sensor checks are not required when these instruments are not required to be OPERABLE or are tripped. In addition, the Note states that if tests are missed, they shall be performed prior to returning the systems to an operable status. These explicit requirements are not retained in ITS 3.3.3.1. This changes the CTS by not including these explicit requirements.

The purpose of this Note is to provide guidance on when Surveillances are required to be met and performed. This explicit Note is not needed in ITS 3.3.3.1 since these allowances are included in ITS SR 3.0.1. ITS SR 3.0.1 states that SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR, and failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. SR 3.0.1 also states that SRs are not required to be performed on inoperable equipment. When equipment is declared inoperable, the Actions of this LCO require the equipment to be placed in the trip condition. In this condition, the equipment is still inoperable but has accomplished the required safety function. Therefore the allowances in SR 3.0.1 and the associated actions provide adequate guidance with respect to when the associated surveillances are required to be performed and this explicit requirement is not retained. This change is administrative because it does not result in a technical change to the CTS.

A.4 CTS Table 4.14.1 specifies an "Operating Cycle" Frequency for the CHANNEL CALIBRATION. ITS SR 3.3.3.1.2 requires performance of a CHANNEL CALIBRATION every "24 months." This changes the CTS by changing the Frequency from once per "Operating Cycle" to "24 months."

DISCUSSION OF CHANGES
ITS 3.3.3.1, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

This change is acceptable because the current "Operating Cycle" is "24 months." In letter L-MT-04-036, from Thomas J. Palmisano (NMC) to the USNRC, dated June 30, 2004, NMC has proposed to extend the fuel cycle from 18 months to 24 months and at the same time has performed an evaluation in accordance with Generic Letter 91-04 to extend the unit Surveillance Requirements from 18 months to 24 months. CTS Table 4.14.1 was included in this evaluation. This change is administrative because it does not result in any technical changes to the CTS.

A.5 CTS Table 4.14.1 Note (3), which applies to the Reactor Vessel Fuel Zone Water Level Monitor states the "Once/month sensor check will consist of verifying that the fuel zone level indicates off scale high." ITS Table 3.3.3.1-1 does not retain this detail. This changes the CTS by deleting a specific method of completing the sensor check.

The CTS Table 4.14.1 Note (3) requirement to verify the fuel zone level indicates off scale is the normal manner in which the CHANNEL CHECK will be performed. The definition of CHANNEL CHECK requires a "qualitative assessment, by observation, of channel behavior during operation." The definition further states "This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter." Therefore the requirement to perform the CHANNEL CHECK (ITS SR 3.3.3.1.1) is sufficient to ensure the channel is functioning properly, and the specific details as to how the CHANNEL CHECK is redundant and unnecessary. This change is administrative because it does not result in any technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.14 is applicable whenever irradiated fuel is in the reactor vessel and reactor water temperature is greater than 212°F. ITS LCO 3.3.3.1 is applicable in MODES 1 and 2. This changes the CTS by requiring PAM instrumentation to be OPERABLE in MODE 2 when reactor water temperature is less than or equal to 212°F.

The purpose of CTS 3.14 is to ensure PAM instrumentation is OPERABLE to display plant variables that provide information required by the control room operators to monitor and diagnose conditions relevant to pre planned actions required to mitigate the consequences of a design basis accident. The PAM instrumentation is required to be OPERABLE during MODES 1 and 2 when the applicable DBAs are assumed to occur. In MODE 1 the reactor coolant temperature will always be above 212°F. In MODE 2, the reactor coolant temperature may be less than or equal to 212°F when the reactor is subcritical but control rods are withdrawn. Therefore, it is necessary and acceptable to require the PAM instrumentation to be OPERABLE. This change is more restrictive because the LCO will be applicable under more reactor operating conditions than in the CTS.

DISCUSSION OF CHANGES
ITS 3.3.3.1, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

M.2 CTS Table 3.14.1 does not require OPERABLE instrument channels for Reactor Vessel Pressure or Penetration Flow Path Primary Containment Isolation Valve (PCIV) Position. These are added to the CTS and specified in ITS Table 3.3.3.1-1, Functions 1 and 6 respectively. Two channels are provided for Reactor Vessel Pressure (Function 1). Two channels per penetration flow path are provided for Penetration Flow Path PCIV Position (Function 6), and is modified by two footnotes, footnotes (a) and (b). Footnote (a) does not require position indication for isolation valves whose penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. Footnote (b) requires only one position indication channel per penetration flow path with one installed channel located in the control room. ITS 3.3.3.1 ACTION A has been added to cover the Condition when one or more Functions have one required channel inoperable, and allows 30 days to restore the required channel to OPERABLE status. If this cannot be met, then ITS 3.3.3.1 ACTION B requires the immediate initiation of the actions specified in Specification 5.6.4. ITS 3.3.3.1 ACTION C has been added to cover the Condition when one or more Functions have two required channels inoperable, and requires restoration of one channel to OPERABLE status within 7 days. If this cannot be met, then ITS 3.3.3.1 ACTION D must be entered, which will then require entry into ACTION E, which requires the plant to be in MODE 3 within 12 hours. A Note has been added to the ACTIONS to allow Separate Condition entry for each Function. Furthermore, SR 3.3.3.1.1 requires a CHANNEL CHECK every 31 days and SR 3.3.3.1.2 requires a CHANNEL CALIBRATION every 24 months for the channels. This changes the CTS by adding new Functions and applicable Footnotes, ACTIONS Note, ACTIONS, and SRs.

This change is acceptable because a plant specific evaluation has concluded that these instrumentation channels are required to provide the primary information to the operator necessary in order to perform manual actions for which no automatic controls exist and that are required for safety systems to accomplish their safety functions for design basis accident (DBA) events. This change is more restrictive because the ITS specifies two Functions, including associated Surveillances and ACTIONS not currently required by the CTS.

RELOCATED SPECIFICATIONS

R.1 CTS Tables 3.14.1 and 4.14.1 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 instruments specified in the plant's

DISCUSSION OF CHANGES
ITS 3.3.3.1, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

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 Monticello Regulatory Guide 1.97 instruments. Those 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 Monticello USAR and the NRC Regulatory Guide 1.97 Safety Evaluation shows that the following Tables 3.14.1 and 4.14.1 Instruments do not meet Category 1 or Type A requirements.

Function 2	Safety/Relief Valve Position
Function 7	Offgas Stack Wide Range Radiation
Function 8	Reactor Building Vent Wide Range Radiation

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

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). These instruments do not meet criterion 1.

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

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

4. These instruments are not structures, systems, or components which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. These instruments do not meet criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met for instruments which do not meet Regulatory Guide 1.97 Type A variable requirements or non-Type A, Category 1, variable requirements, their associated LCO and Surveillances may be relocated out of the Technical Specifications. The Technical Specification requirements for these instruments will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as a relocation because the LCO requirements for these instruments did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and have been relocated to the TRM.

<u>REMOVED DETAIL CHANGES</u>

LA.1 *(Type 3 – Removing Procedural Details for meeting TS Requirements or Reporting Requirements)* CTS Table 3.14.1 Required Condition D requires immediate initiation of the preplanned alternate method of monitoring the appropriate parameters if the number of OPERABLE channels is less than the minimum number of channels (i.e., both of the channels are inoperable). The

DISCUSSION OF CHANGES
ITS 3.3.3.1, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

ITS does not include this requirement. This changes the CTS by moving this detail to the ITS Bases.

The removal of these details for performing Required Actions from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS 3.3.3.1 ACTION F requires action to be immediately initiated in accordance with ITS 5.6.4. ITS 5.6.4 requires a report to be submitted to the NRC within the following 14 days and that the report outline the preplanned alternate method of monitoring. As such, the relocated details are not needed to be included in the ITS. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L.1 *(Category 2 – Relaxation of Applicability)* CTS 3.14 is applicable whenever irradiated fuel is in the reactor vessel and reactor water temperature is greater than 212°F. Consistent with this Applicability, CTS Table 3.14.1 Required Condition B, requires a shutdown to Cold Shutdown (MODE 4) to place the unit outside the Applicability of CTS 3.14. ITS LCO 3.3.3.1 is applicable in MODES 1 and 2. Consistent with this new Applicability, ITS 3.3.3.1 ACTION E only requires a unit shutdown to MODE 3. This changes the CTS by not requiring PAM instrumentation to be OPERABLE in MODE 3 (i.e., reactor water temperature above 212°F and, consistent with this Applicability, only requiring the unit to be shut down to MODE 3 instead of to MODE 4.

The purpose of CTS 3.14 is to ensure PAM instrumentation is OPERABLE to display plant variables that provide information required by the control room operators to monitor and diagnose conditions relevant to pre planned actions required to mitigate the consequences of a design basis accident. The PAM instrumentation is required to be OPERABLE during MODES 1 and 2 when the applicable DBAs are assumed to occur. These instruments should not be required in MODE 3 because they are required to monitor variables related to the diagnosis and preplanned actions required to mitigate design basis accidents occurring in MODES 1 and 2. In MODE 3, plant conditions are such that the likelihood of an event that would require PAM instrumentation is extremely low. Therefore, this change to not require the PAM instrumentation to be OPERABLE in MODE 3 is acceptable. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

L.2 *(Category 4 – Relaxation of Required Action)* CTS 4.14 does not provide a delayed entry into associated Conditions and Required Actions if a PAM channel

DISCUSSION OF CHANGES
ITS 3.3.3.1, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

is inoperable solely for performance of required Surveillances. ITS 3.3.3.1 Surveillance Requirements Note 2 has been added to allow delayed entry into associated Conditions and Required Actions for up to 6 hours if a PAM channel is placed in an inoperable status solely for performance of required Surveillances provided the associated Function maintains capability. This changes the CTS by providing a delay time to enter Conditions and Required Actions for a PAM channel placed in an inoperable status solely for performance of required Surveillances.

ITS 3.3.3.1 Surveillance Requirements Note 2 has been added to allow delayed entry into associated Conditions and Required Actions for up to 6 hours if a PAM channel is placed in an inoperable status solely for performance of required Surveillances, provided the other required channel in the associated Function is OPERABLE. This change is acceptable because it provides a reasonable time for performing tests and reduces the risk of error during testing. This change is acceptable since the 6 hour testing allowance does not significantly reduce the probability of properly monitoring post accident parameters, when necessary, since the other channel monitoring the variable must be OPERABLE for this allowance to be used. This allowance has been granted by the NRC during the conversion to the ITS for WNP 2, Nine Mile Point Unit 2, Quad Cities Units 1 and 2, Dresden Units 2 and 3, LaSalle Units 1 and 2, and FitzPatrick. Also the NRC has granted this allowance for other equipment in accordance with NEDC-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988. This change is less restrictive because less stringent Required Actions are being applied in ITS than were applied in CTS.

L.3 *(Category 3 – Relaxation of Completion Time)* CTS Table 3.14.1 Required Condition A allows 7 days to restore an inoperable PAM channel when the number of OPERABLE channels is one less than the total number of channels (i.e., one of the two channels inoperable). ITS 3.3.3.1 ACTION A allows 30 days to restore an inoperable required channel to OPERABLE status when one of the two channels of a PAM Function is inoperable. This changes the CTS by extending the time to restore an inoperable PAM instrumentation channel from 7 days to 30 days.

The purpose of CTS Table 3.14.1 Required Condition A is to allow time to restore an inoperable PAM instrument channel. This change is acceptable because the Completion Time of 30 days is based on operating experience and takes into account the remaining OPERABLE channels, the passive nature of the instruments, and the relatively low probability of an event requiring PAM instrument operation during this interval. This change is less restrictive because more time is allowed to restore an inoperable PAM instrument channel to OPERABLE status in the ITS than was allowed in the CTS.

L.4 *(Category 3 – Relaxation of Completion Time)* CTS Table 3.14.1 Required Condition B allows 48 hours to restore an inoperable PAM channel when the number of OPERABLE channels is less than the minimum number of channels (i.e., both of the channels are inoperable). This Required Condition applies to the Reactor Vessel Fuel Zone Water Level, Drywell Wide Range Pressure, and Suppression Pool Wide Range Level PAM channels. ITS 3.3.3.1 ACTION C allows 7 days to restore one required inoperable channel to OPERABLE status

DISCUSSION OF CHANGES
ITS 3.3.3.1, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

when both of the channels of a PAM Function are inoperable. This changes the CTS by extending the time to restore an inoperable PAM instrumentation channel, when two channels are inoperable in the same Function, from 48 hours to 7 days.

The purpose of CTS Table 3.14.1 Required Condition B is to allow time to restore one inoperable PAM instrument channel, when two channels are inoperable in the same Function, before requiring a reactor shutdown. This change is acceptable because the Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain required information. This change is less restrictive because more time is allowed to restore an inoperable PAM instrument channel to OPERABLE status in the ITS than was allowed in the CTS.

L.5 *(Category 3 – Relaxation of Completion Time)* CTS Table 3.14.1 Required Condition D requires immediate initiation of the preplanned alternate method of monitoring the appropriate parameters and the submittal of the report required by Required Condition A if the number of OPERABLE channels is less than the minimum number of channels (i.e., both of the channels are inoperable). This Required Condition applies to the Suppression Pool Temperature and Drywell High Range Radiation PAM channels. ITS 3.3.3.1 ACTION C allows 7 days to restore one required inoperable channel to OPERABLE status when both of the channels of a PAM Function are inoperable. This changes the CTS by providing a 7 day restoration time when two channels are inoperable in the same Function prior to requiring the submittal of a report.

The purpose of CTS Table 3.14.1 Required Condition D is to provide compensatory actions when two channels are inoperable in the same Function, before requiring a reactor shutdown. This change is acceptable because the Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain required information. This change is less restrictive because more time is allowed to restore an inoperable PAM instrument channel to OPERABLE status in the ITS than was allowed in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.3 INSTRUMENTATION

3.3.3.1 Post Accident Monitoring (PAM) Instrumentation

3.14 LCO 3.3.3.1 The PAM instrumentation for each Function in Table 3.3.3.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----

DOC A.2 Separate Condition entry is allowed for each Function.

	CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.14.1 Required Condition A	A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
Table 3.14.1 Required Condition A	B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.7. 5 4	Immediately TSTF-369 8
Table 3.14.1 Required Condition B	C. One or more Functions with two required channels inoperable.	C.1 Restore one required channel to OPERABLE status.	7 days
Table 3.14.1 Required Conditions B and D	D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.3.1-1 for the channel.	Immediately

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME	
Table 3.14.1 Required Condition B	E. As required by Required Action D.1 and referenced in Table 3.3.3.1-1.	E.1 Be in MODE 3.	12 hours	
Table 3.14.1 Required Condition D	F. [As required by Required Action D.1 and referenced in Table 3.3.3.1-1.	F.1 Initiate action in accordance with Specification 5.6.~~7~~ 5 4.	Immediately]	1 TSTF-369 8

SURVEILLANCE REQUIREMENTS

4.14 1. ---NOTE S-- 6
These SRs apply to each Function in Table 3.3.3.1-1.

INSERT 1

CTS	SURVEILLANCE		FREQUENCY	
Table 4.14.1	SR 3.3.3.1.1	Perform CHANNEL CHECK.	31 days	
Table 4.14.1	SR 3.3.3.1.2	Perform CHANNEL CALIBRATION.	~~[18]~~ 24 months	1

CTS

6 **INSERT 1**

DOC L.2

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the other required channel in the associated Function is OPERABLE.

CTS

Table 3.3.3.1-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

CTS	FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	
Table 3.14.1				
DOC M.2	1. Reactor ~~Steam Dome~~ Vessel Pressure	2	E	2
1	2. Reactor Vessel Water Level	2	E	
4	3. Suppression Pool Water Level	2	E	
3	4. Drywell Pressure	2	E	
6	5. Primary Containment Area Radiation	2	[F]	1
	~~[6. Drywell Sump Level~~	~~2~~	~~E]~~	3
	~~[7. Drywell Drain Sump Level~~	~~2~~	~~E]~~	3
DOC M.2	~~8~~ 6. Penetration Flow Path PCIV Position	2 per penetration flow path[a] [b]	E	3
	~~9. Wide Range Neutron Flux~~	~~2~~	~~E~~	4
	~~10. Primary Containment Pressure~~	~~2~~	~~E~~	4
5	~~11~~ 7. ~~[Relief Valve Discharge Location]~~ Suppression Pool Water Temperature	2~~[c]~~	~~E~~ F	4 1 7

DOC M.2 (a) Not required for isolation valves whose associated penetration flow path is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

DOC M.2 (b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

~~(c) Monitoring each [relief valve discharge location].~~ 1

~~REVIEWER'S NOTE~~
~~Table 3.3.3.1-1 shall be amended for each plant as necessary to list:~~
~~1. All Regulatory Guide 1.97, Type A instruments and~~
~~2. All Regulatory Guide 1.97, Category 1, non-Type A instruments specified in the plant's Regulatory Guide 1.97, Safety Evaluation Report.~~ 5

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.3.1, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. The bracketed items have been deleted since they are not applicable to Monticello. Subsequent Functions have been renumbered to reflect this change.

4. ISTS Table 3.3.3.1-1 Function 9, Wide Range Neutron Flux, has been deleted. This Function is not retained in the PAM instrumentation as provided in NRC (B.A. Wetzel) letter to NSP (R.O. Anderson), "Regulatory Guide 1.97 – Boiling Water Reactor Neutron Flux Monitoring – Monticello Nuclear Generating Plant," dated February 24, 1994. ISTS Table 3.3.3.1-1 Function 10, Primary Containment Pressure, has been deleted since it is not a Type A or Category 1 variable at Monticello. Subsequent Functions have been renumbered to reflect this change.

5. This Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

6. ITS Surveillance Requirements Note 2 has been added to allow delayed entry into associated Conditions and Required Actions for up to 6 hours if a PAM channel is placed in an inoperable status for performance of required Surveillances provided the other required channel in the associated Function is OPERABLE. This allowance has been granted by the NRC during the conversion to the ITS for WNP 2, Nine Mile Point Unit 2, Quad Cities Units 1 and 2, Dresden Units 2 and 3, LaSalle Units 1 and 2, and FitzPatrick. Also the NRC has granted this allowance for other equipment in accordance with NEDC-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988. In addition the current Note to the Surveillance Requirements for ITS 3.3.3.1 has been renumbered "1" to reflect this addition.

7. The required Condition to enter when two Suppression Pool Water Temperature channels are inoperable and one is not restored within 7 days (as required by ACTION C) has been changed from Condition E to Condition F. This will allow unit operation to continue provided action is initiated in accordance with Specification 5.6.4. This is also consistent with the current Technical Specifications (CTS Table 3.14.1 Required Condition D).

8. Changes have been made to be consistent with changes made in another Specification.

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Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION

B 3.3.3.1 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Events. The instruments that monitor these variables are designated as Type A, Category I, and non-Type A, Category I, in accordance with Regulatory Guide 1.97 (Ref. 1). 1

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident. This capability is consistent with the recommendations of Reference 1.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation LCO ensures the OPERABILITY of Regulatory Guide 1.97, Type A variables so that the control room operating staff can:

- Perform the diagnosis specified in the Emergency Operating Procedures (EOPs). These variables are restricted to preplanned actions for the primary success path of Design Basis Accidents (DBAs), (e.g., loss of coolant accident (LOCA)), and 2
- Take the specified, preplanned, manually controlled actions for which no automatic control is provided, which are required for safety systems to accomplish their safety function.

The PAM instrumentation LCO also ensures OPERABILITY of Category I, non-Type A, variables so that the control room operating staff can: 1

- Determine whether systems important to safety are performing their intended functions, 2
- Determine the potential for causing a gross breach of the barriers to radioactivity release, 2
- Determine whether a gross breach of a barrier has occurred, and 2
- Initiate action necessary to protect the public and for an estimate of the magnitude of any impending threat.

BASES

APPLICABLE SAFETY ANALYSES (continued)

conformance requirements 3

The plant specific Regulatory Guide 1.97 Analysis (Ref. 2) documents the process that identified Type A and Category ~~I~~ 1, non-Type A, variables. 1

Accident monitoring instrumentation that satisfies the definition of Type A in Regulatory Guide 1.97 meets Criterion 3 of 10 CFR 50.36(c)(2)(ii). Category ~~I~~ 1, non-Type A, instrumentation is retained in Technical Specifications (TS) because they are intended to assist operators in minimizing the consequences of accidents. Therefore, these Category ~~I~~ 1 variables are important for reducing public risk. 1 1

LCO

LCO 3.3.3.1 requires two OPERABLE channels for all but one Function to ensure that no single failure prevents the operators from being presented with the information necessary to determine the status of the plant and to bring the plant to, and maintain it in, a safe condition following ~~that~~ an accident. 1

Furthermore, ~~provision of~~ providing two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information. ~~[More than two channels may be required at some plants if the Regulatory Guide 1.97 analysis determined that failure of one accident monitoring channel results in information ambiguity (that is, the redundant displays disagree) that could lead operators to defeat or to fail to accomplish a required safety function.]~~ 2 4

The exception to the two channel requirement is primary containment isolation valve (PCIV) position. In this case, the important information is the status of the primary containment penetrations. The LCO requires one position indicator for each active PCIV. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of passive valve or via system boundary status. If a normally active PCIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

The following list is a discussion of the specified instrument Functions listed in Table 3.3.3.1-1 in the accompanying LCO. ~~These discussions are intended as examples of what should be provided for each Function when the plant specific list is prepared.~~ 4

BASES

LCO (continued)

Vessel

1. Reactor Steam Dome Pressure (6)

vessel

1

Reactor steam dome pressure is a Category I variable provided to support monitoring of Reactor Coolant System (RCS) integrity and to verify operation of the Emergency Core Cooling Systems (ECCS). Two independent pressure transmitters with a range of 0 psig to 1500 psig monitor pressure. Wide range recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel. (3) (1) (3)

indicators

2. Reactor Vessel Water Level

Type A and

1

Reactor vessel water level is a Category I variable provided to support monitoring of core cooling and to verify operation of the ECCS. The wide range water level channels provide the PAM Reactor Vessel Water Level Function. The wide range water level channels measure from 17 inches below the dryer skirt down to a point just below the bottom of the active fuel. Wide range water level is measured by two independent differential pressure transmitters. The output from these channels is recorded on two independent pen recorders, which is the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel. (1) (3) (3) (3)

reactor vessel fuel zone

INSERT 1

Reactor vessel fuel zone

INSERT 2

reactor vessel fuel zone

The wide range water level instruments are uncompensated for variation in reactor water density and are calibrated to be most accurate at operational pressure and temperature.

3. Suppression Pool Water Level

1

Suppression pool water level is a Category I variable provided to detect a breach in the reactor coolant pressure boundary (RCPB). This variable is also used to verify and provide long term surveillance of ECCS function. The wide range suppression pool water level measurement provides the operator with sufficient information to assess the status of both the RCPB and the water supply to the ECCS. The wide range water level indicators monitor the suppression pool water level from the center line of the ECCS suction lines to the top of the pool. Two wide range suppression pool water level signals are transmitted from separate differential pressure transmitters and are continuously recorded on two recorders in the control room. These recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel. (1) (3)

recorders

-8 ft to +15 ft, with 0 inches corresponding to the 910 ft elevation

3 **INSERT 1**

provide indication, based on instrument zero, from -335 inches to +65 inches, which includes the reactor vessel fuel zone and normal operating range.

3 **INSERT 2**

One reactor vessel fuel zone wide range channel consists of a transmitter and a control room indicator. The other reactor vessel fuel zone wide range channel consists of a transmitter, a control room indicator, and a control room recorder.

BASES

LCO (continued)

4. Drywell Pressure

1

Drywell pressure is a Category ~~I~~ variable provided to detect breach of the RCPB and to verify ECCS functions that operate to maintain RCS integrity. Two wide range drywell pressure signals are transmitted from separate pressure transmitters and are continuously recorded and displayed on two control room recorders. These recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel. 1

5. Primary Containment Area Radiation ~~(High Range)~~ 8

Primary containment area radiation (high range) is provided to monitor the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. ~~[For this plant,]~~ primary containment area radiation (high range) PAM instrumentation consists of ~~the following:]~~ 5

INSERT 3

~~6. Drywell Sump Level~~

~~Drywell sump level is a Category I variable provided for verification of ECCS functions that operate to maintain RCS integrity. [For this plant, the drywell sump level PAM instrumentation consists of the following:]~~ 6

~~7. Drywell Drain Sump Level~~

~~Drywell drain sump level is a Category I variable provided to detect breach of the RCPB and for verification and long term surveillance of ECCS functions that operate to maintain RCS integrity. [For this plant, the drywell drain sump level PAM instrumentation consists of the following:]~~ 6

6

Penetration Flow Path

~~8~~. Primary Containment Isolation Valve (PCIV) Position 6 8

primary

PCIV position is provided for verification of containment integrity. In the case of PCIV position, the important information is the isolation status of the containment penetration. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active PCIV in a containment penetration flow path, i.e., two total channels of PCIV position indication for a penetration flow path with two active valves. For containment penetrations with only one active PCIV having control room indication, Note (b) requires a single channel of valve position 3 3 3 3

5 **INSERT 3**

two physically separated and redundant radiation detectors with a range of 1 R/hr to 10 E8 R/hr located inside the drywell. The detectors provide a signal to separate radiation monitor recorders located in the control room. These detectors and associated recorders in the control room provide the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with these portions of the instrument channel.

BASES

LCO (continued)

indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration via indicated status of the active valve, as applicable, and prior knowledge of passive valve or system boundary status. If a penetration flow path is isolated, position indication for the PCIV(s) in the associated penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration flow path is not required to be OPERABLE. Each penetration is treated separately and each penetration flow path is considered a separate function. Therefore, separate Condition entry is allowed for each inoperable penetration flow path. 1

Penetration Flow Path 8

INSERT 4

~~[For this plant,]~~ ~~the~~ PCIV position PAM instrumentation consists of ~~[the following:]~~ 5

~~9. Wide Range Neutron Flux~~

~~Wide range neutron flux is a Category I variable provided to verify reactor shutdown. [For this plant, the wide range neutron flux PAM instrumentation consists of the following:]~~ 6

~~10. Primary Containment Pressure~~

~~Primary containment pressure is a Category I variable provided to verify RCS and containment integrity and to verify the effectiveness of ECCS actions taken to prevent containment breach. Two wide range primary containment pressure signals are transmitted from separate pressure transmitters and are continuously recorded and displayed on two control room recorders. These recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.~~ 6

7 ~~11.~~ Suppression Pool Water Temperature 6

Type A and 1

Suppression pool water temperature is ~~a~~ Category ~~I~~ variable provided to detect a condition that could potentially lead to containment breach and to verify the effectiveness of ECCS actions taken to prevent containment breach. The suppression pool water temperature instrumentation allows operators to detect trends in suppression pool water temperature ~~in sufficient time to take action to prevent steam quenching vibrations in the suppression pool. Twenty-four temperature sensors are arranged in six groups of four independent and redundant channels, located such that there is a group of sensors within a 30 ft line of sight of each relief valve discharge location.~~ 1 3

INSERT 5

5 **INSERT 4**

position switches mounted on the valves for the positions to be indicated, associated wiring, and control room indicating lamps for active PCIVs (check valves and manual valves are not required to have position indication). These position switches and associated indicators in the control room provide the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with these portions of the instrument channel.

5 **INSERT 5**

. The suppression pool water temperature is monitored by two redundant channels. Each channel consists of eight resistance temperature detectors (RTDs) that monitor temperature over a range of 30°F to 230°F. The RTDs are mounted in thermowells spaced at equal intervals around the periphery of the suppression pool. The eight RTD signals are averaged and the resulting bulk temperature is sent to redundant indicating recorders in the control room.

BASES

LCO (continued)

~~Thus, six groups of sensors are sufficient to monitor each relief valve discharge location. Each group of four sensors includes two sensors for normal suppression pool temperature monitoring and two sensors for PAM. The outputs for the PAM sensors are recorded on four independent recorders in the control room (channels A and C are redundant to channels B and D, respectively). All four of these recorders must be OPERABLE to furnish two channels of PAM indication for each of the relief valve discharge locations.~~ These recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channels. ~~Each suppression pool water temperature [relief valve discharge location] is treated separately and each [relief valve discharge location] is considered to be a separate function. Therefore, separate Condition entry is allowed for each inoperable [relief valve discharge location.]~~ 3 6

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1 and 2. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1 and 2. In MODES 3, 4, and 5, plant conditions are such that the likelihood of an event that would require PAM instrumentation is extremely low; therefore, PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

A Note has been provided to modify the ACTIONS related to PAM instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable PAM instrumentation channels provide appropriate compensatory measures for separate Functions. As such, a Note has been provided that allows separate Condition entry for each inoperable PAM Function.

BASES

ACTIONS (continued)

A.1

When one or more Functions have one required channel that is inoperable, the required inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channels (or, in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

If a channel has not been restored to OPERABLE status in 30 days, this Required Action specifies initiation of action in accordance with Specification 5.6.7, which requires a written report to be submitted to the NRC. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement, since alternative actions are identified before loss of functional capability, and given the likelihood of plant conditions that would require information provided by this instrumentation.

C.1

When one or more Functions have two required channels that are inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

BASES

ACTIONS (continued)

D.1

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.3.1-1. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met the Required Action of Condition C and the associated Completion Time has expired, Condition D is entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1

For the majority of Functions in Table 3.3.3.1-1, if the Required Action and associated Completion Time of Condition C is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

and suppression pool water temperature

Since alternate means of monitoring primary containment area radiation have been developed and tested, the Required Action is not to shut down the plant, but rather to follow the directions of Specification 5.6.~~7~~. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels. 6 TSTF-369 5 4 6

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, ~~T~~the following SRs apply to each PAM instrumentation Function in Table 3.3.3.1-1. 7

INSERT 6 6

SR 3.3.3.1.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel against a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter

6 **INSERT 6**

The Surveillances are modified by a second Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. The 6 hour testing allowance is acceptable since it does not significantly reduce the probability of properly monitoring post accident parameters when necessary.

BASES

SURVEILLANCE REQUIREMENTS (continued)

should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to ~~similar plant~~ other radiation instruments located throughout the plant. 3

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency of 31 days is based upon plant operating experience, with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 31 day interval is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of those displays associated with the required channels of this LCO.

SR 3.3.3.1.2 8

A CHANNEL CALIBRATION is required to be performed every ~~18~~ 24 months~~, or approximately at every refueling~~. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies the channel responds to measured parameter with the necessary range and accuracy. 6

The Frequency is based on operating experience and consistency with the typical industry refueling cycles.

REFERENCES

1. Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," ~~[date]~~ Revision 3, May 1983 5

2. ~~[Plant specific documents (e.g., NRC Regulatory Guide 1.97, SER letter).]~~ USAR, Section 7.9.3. 5

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.3.1 BASES, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

1. Typographical/grammatical error corrected.
2. These punctuation corrections have been made consistent with the Writers Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.
3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.
4. This Reviewer's Note (or reviewer's type of note) has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.
5. The brackets have been removed and the proper plant specific information/value has been provided.
6. Changes have been made to reflect those changes made to the Specification.
7. Editorial changes made to be consistent with similar statements in other places in the Bases.
8. Changes have been made to more closely reflect the Specification requirements.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS ITS 3.3.3.1, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

There are no specific NSHC discussions for this Specification.

ATTACHMENT 6

ITS 3.3.3.2, Alternate Shutdown System

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS | ITS

3.0 LIMITING CONDITIONS FOR OPERATION

3.13 ALTERNATE SHUTDOWN SYSTEM

~~Applicability:~~

~~Applies to system controls on the alternate shutdown panel.~~

~~Objective:~~

~~To insure the integrity of the alternate shutdown system.~~

Specification:

3.3.3.2 A. Alternate Shutdown System

LCO 3.3.3.2 / Applicability 1. The system controls on the ASDS panel shall be operable ~~whenever that system/component is required to be operable.~~ ~~12 RHR Service Water Pump shall be operable from the ASDS panel whenever there is irradiated fuel in the vessel and reactor water temperature is greater than 212°F.~~

MODES 1 and 2 — M.1, L.1

A.2

ACTION A 2. If system controls ~~or 12 RHR Service Water Pump~~ required to be operable by Specification 3.13.A.1 are made or found inoperable, restore operability within ~~7~~ 30 days, or perform one of the following: — L.2

~~a. Provide equivalent shutdown capability and within 60 days restore the inoperable system controls to operable; or~~

~~b. Establish a continuous fire watch in the cable spreading room and the back-panel area of the control room and within 60 days restore the inoperable system controls to operable; or~~

M.2

Add proposed ACTIONS Note — A.4

4.0 SURVEILLANCE REQUIREMENTS

4.13 ALTERNATE SHUTDOWN SYSTEM

~~Applicability:~~

~~Applies to the periodic testing of controls on the alternate shutdown system panel.~~

~~Objective:~~

~~To verify the operability of controls on the alternate shutdown panel.~~

Specification:

A. Alternate Shutdown System

SR 3.3.3.2.2 1. Switches on the alternate shutdown system panel shall be functionally tested once per ~~operating cycle.~~ 24 months — A.3

SR 3.3.3.2.2 2. The alternate shutdown system panel master transfer switch shall be verified ~~to alarm in the control room when unlocked~~ once per ~~operating cycle.~~ — LA.1, A.3

Add proposed SR 3.3.3.2.1 and SR 3.3.3.2.3 — M.3

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A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
c. ~~Verify the operability of the fire detectors in the cable spreading room and the back-panel area of the control room and establish a hourly fire watch patrol and within 60 days restore the inoperable system controls to operable; or~~	M.2
ACTION B — d. Place the reactor in ~~a condition where the systems for which the system controls at the ASDS are inoperable are not required to be operable~~ within ~~24~~ hours.	MODE 3; 12; M.1; L.1
LCO 3.3.3.2 — 3. ~~The alternate shutdown system panel master transfer switch shall be locked in the normal position except when in use, being tested or being maintained.~~	LA.2

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DISCUSSION OF CHANGES
ITS 3.3.3.2, ALTERNATE SHUTDOWN SYSTEM

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 3.13.A.1 states, in part, that 12 RHR service water pump shall be OPERABLE whenever there is irradiated fuel in the vessel and water temperature is greater than 212 F. CTS 3.13.A.2 states, in part, that an inoperable 12 RHR service water pump shall be restored to OPERABLE within 7 days. ITS LCO 3.3.3.2 retains requirements for OPERABILITY of the 12 RHR service water controls associated with the Alternate Shutdown System but does not retain requirements for the OPERABILITY of the 12 RHR service water pump. This changes the CTS by deleting 12 RHR service water pump OPERABILITY requirements from the Alternate Shutdown System Specification.

This change is acceptable because a new Specification, ITS LCO 3.7.1, "RHR Service Water System," has been added (ITS 3.7.1 DOC M.1) to address the OPERABILITY of the RHR Service Water System, including 12 RHR service water pump. Therefore, the requirements established in CTS 3.13.A.1 and CTS 3.13.A.2 for 12 RHR service water pump are redundant and are not required. In addition, the CTS 3.13 requirements for 12 RHR Service water pump were added by CTS Amendment 113, October 2, 2000 to meet the intent of GL 81-12 to ensure any equipment not covered by an existing TS were addressed by the Alternate Shutdown System. This change is administrative because it does not result in technical changes to the CTS.

A.3 CTS 4.13.A.1 requires the switches on the Alternate Shutdown System be functionally tested once per "operating cycle." CTS 4.13.A.2 requires the Alternate Shutdown System panel master transfer switch to be functionally tested once per "operating cycle." ITS SR 3.3.3.2.2 requires performance of a similar test every "24 months." This changes the CTS by changing the Frequency from once per "operating cycle" to "24 months."

This change is acceptable because the current "operating cycle" is "24 months." In letter L-MT-04-036, from Thomas J. Palmisano (NMC) to the USNRC, dated June 30, 2004, NMC has proposed to extend the fuel cycle from 18 months to 24 months and at the same time has performed an evaluation in accordance with Generic Letter 91-04 to extend the unit Surveillance Requirements from 18 months to 24 months. CTS 4.13.A.1 and 4.13.A.2 were included in this evaluation. This change is administrative because it does not result in any technical changes to the CTS.

A.4 CTS 3.13.A.1 specifies the compensatory actions to take when Alternate Shutdown System controls are inoperable. ITS 3.3.3.2 ACTIONS provide the compensatory actions for inoperable Alternate Shutdown System Functions. The

DISCUSSION OF CHANGES
ITS 3.3.3.2, ALTERNATE SHUTDOWN SYSTEM

ITS 3.3.3.2 ACTIONS include a Note that allows separate Condition entry for each Function. This modifies the CTS by providing a specific allowance to enter the Action for each inoperable Alternate Shutdown System Function.

This change is acceptable because it clearly states the current requirement. The CTS considers each Alternate Shutdown System Function to be separate and independent from the others. This change is administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.13.A.1 states that the ASDS system controls on the ASDS panel shall be OPERABLE "whenever that system or component is required to be OPERABLE." For the system and components covered by this Specification, the Applicability that covers the most conditions is whenever irradiated fuel is in the reactor vessel and the reactor water temperature is greater than 212°F (i.e., the RHR pumps Applicability). In addition, when the restoration time provided by CTS 3.13.A.2.b has expired, CTS 3.13.A.2.d requires placing the reactor in a condition where the systems for which the system controls at the ASDS are inoperable are not required to be OPERABLE in 24 hours. ITS LCO 3.3.3.2 is applicable in MODES 1 and 2. Consistent with this Applicability change, ITS 3.3.3.2 ACTION B requires the plant to be in MODE 3 within 12 hours. This changes the CTS by requiring Alternate Shutdown System controls and instrumentation to be OPERABLE in MODE 2 when reactor water temperature is ≤ 212°F and provides only 12 hours in lieu of 24 hours to exit the Applicability.

The purpose of CTS 3.13.A.1 is to ensure the Alternate Shutdown System is OPERABLE to provide the control room operators with sufficient controls and instrumentation to promptly shutdown the reactor and maintain the plant in a safe condition at a location other than the control room. ITS LCO 3.3.3.2 is applicable in MODES 1 and 2 so that the plant can be placed in, and maintained in, MODE 3 for an extended period of time from a location other than the control room. In MODE 1 the reactor coolant temperature will always be above 212°F. In MODE 2, the reactor coolant temperature may be less than or equal to 212°F when the reactor is subcritical but control rods are withdrawn. Therefore, it is necessary and acceptable to require the Alternate Shutdown System controls and instrumentation to be OPERABLE. Furthermore, 12 hours is sufficient to reach MODE 3 from MODE 1. This change is more restrictive because the LCO will be applicable under more reactor operating conditions than in the CTS.

M.2 CTS 3.13.A.2.a, b, and c, provide alternative actions and an allowance of up to 60 days before requiring a reactor shutdown if an inoperable Alternate Shutdown System control cannot be restored to an OPERABLE status within 7 days. ITS 3.3.3.2 does not provide these alternative Required Action allowances. This changes the CTS by deleting alternative actions and extended time allowances for inoperable Alternate Shutdown System controls.

The purpose of CTS 3.13.A.2.a, b, and c, is to provide alternative actions and an allowance of up to 60 days before requiring a reactor shutdown if an inoperable

DISCUSSION OF CHANGES
ITS 3.3.3.2, ALTERNATE SHUTDOWN SYSTEM

Alternate Shutdown System control cannot be restored to an OPERABLE status within 7 days. ITS 3.3.3.2 Required Action A Completion Time allows 30 days (see DOC L.2) to restore an inoperable Alternate Shutdown System Function to OPERABLE status. This change is acceptable because the ITS 3.3.3.2 Required Action A still requires restoration of the inoperable Function and the Completion Time of 30 days is reasonable based on operating experience and the low probability of an event that would require evacuation of the control room. This change is more restrictive because alternative actions and extended Completion Times are not allowed in the ITS that were allowed in the CTS.

M.3 CTS 4.13.A does not specify a CHANNEL CHECK or CHANNEL CALIBRATION for required instrumentation channels. ITS SR 3.3.3.2.1 requires a CHANNEL CHECK of each required instrumentation channel that is normally energized every 31 days, and ITS SR 3.3.3.2.3 requires a CHANNEL CALIBRATION of each required instrumentation channel every 24 months. This changes the CTS by adding Surveillance Requirements that were not previously required.

The purpose of CTS 4.13.A is to specify the tests and calibrations required to ensure the Alternate Shutdown System is capable of performing its function. Similarly, ITS 3.3.3.2 includes those Surveillance Requirements required by CTS 4.13.A but also specifies additional Surveillance Requirements for the Alternate Shutdown System instrumentation. These Surveillance Requirements include ITS SR 3.3.3.2.1, a CHANNEL CHECK of each required instrumentation channel that is normally energized every 31 days, and ITS SR 3.3.3.2.3, a CHANNEL CALIBRATION of each required instrumentation channel every 24 months. This change is acceptable because it ensures the Alternate Shutdown System instrumentation will perform their intended function. The proposed Frequency of 31 days for ITS SR 3.3.3.2.1 is based on operating experience that demonstrates channel failure is rare. The proposed ITS SR 3.3.3.2.3 Frequency of 24 months is based on operating experience and consistency with the refueling cycle. This change is more restrictive because the ITS specifies Surveillance Requirements not currently required by the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for meeting TS Requirements or reporting Requirements)* CTS 4.13.A.2 requires verification that the Alternate Shutdown System panel master transfer switch alarms in the control room when unlocked once per operating cycle. ITS SR 3.3.3.2.2 does not include the specifics of how to functionally test the Alternate Shutdown System panel master transfer switch (i.e., verifies it alarms in the control room when unlocked). This changes the CTS by relocating this specific detail to the ITS Bases.

The removal of the detail for performing a Surveillance Requirement from the Technical Specifications is acceptable because this type of information is not

DISCUSSION OF CHANGES
ITS 3.3.3.2, ALTERNATE SHUTDOWN SYSTEM

necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. ITS SR 3.3.3.2.2 requires verification that each control circuit and transfer switch is capable of performing the intended function. This includes the master transfer switch and its associated alarm. As such, the relocated details are not needed to be included in the ITS. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is a less restrictive removal of detail change because procedural details for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.2 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 3.13.A.3 requires the Alternate Shutdown System panel master transfer switch to be locked in the normal position, except when in use, being tested, or being maintained. ITS 3.3.3.2 does not retain this information. This changes the CTS by moving the specific conditions of Alternate Shutdown System OPERABILITY to the ITS Bases.

The removal of these details, which are related to system design, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the Alternate Shutdown System to be OPERABLE (ITS LCO 3.3.3.2). Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

<u>LESS RESTRICTIVE CHANGES</u>

L.1 *(Category 2 – Relaxation of Applicability)* CTS 3.13.A.1 states that the Alternate Shutdown System controls on the ASDS panel shall be OPERABLE whenever that system or component is required to be OPERABLE. For the system and components covered by this Specification, the Applicability that covers the most conditions is whenever irradiated fuel is in the reactor vessel and the reactor water temperature is greater than 212°F (i.e., the RHR pumps Applicability). In addition, when the restoration time provided by CTS 3.13.A.2.b has expired, CTS 3.13.A.2.d requires placing the reactor in a condition where the systems for which the system controls at the ASDS are inoperable are not required to be OPERABLE in 24 hours. ITS LCO 3.3.3.2 is applicable in MODES 1 and 2. Consistent with this Applicability change, ITS 3.3.3.2 ACTION B requires the plant to be in MODE 3 within 12 hours. This changes the CTS by not requiring the Alternate Shutdown System to be OPERABLE in MODE 3 (i.e., reactor water

DISCUSSION OF CHANGES
ITS 3.3.3.2, ALTERNATE SHUTDOWN SYSTEM

temperature above 212°F and, consistent with this Applicability, only requiring the unit to be shut down to MODE 3 instead of to MODE 4.

The purpose of CTS 3.13.A.1 is to ensure the Alternate Shutdown System is OPERABLE to provide the control room operators with sufficient controls and instrumentation to promptly shutdown the reactor and maintain the plant in a safe condition at a location other than the control room. ITS LCO 3.3.3.2 is applicable in MODES 1 and 2 so that the plant can be placed in, and maintained in, MODE 3 for an extended period of time from a location other than the control room. This change is acceptable since the Applicability of MODES 1 and 2 ensure the Alternate Shutdown System Functions are OPERABLE and the plant can be placed in, and maintained in, MODE 3 for an extended period of time from a location other than the control room. This change is less restrictive because the LCO will be applicable under fewer reactor operating conditions than in the CTS.

L.2 *(Category 3 – Relaxation of Completion Time)* CTS 3.13.A.2 allows 7 days to restore an inoperable Alternate Shutdown System Function. ITS 3.3.3.2 ACTION A allows 30 days to restore one required inoperable Function to OPERABLE status. This changes the CTS by extending the time to restore an inoperable Alternate Shutdown System Function from 7 days to 30 days.

The purpose of CTS 3.13.A.2 is to allow time to restore one inoperable Alternate Shutdown System Function before requiring a reactor shutdown. This change is acceptable because the Completion Time of 30 days is based on the relatively low probability of an event requiring evacuation of the control room. This change is less restrictive because more time is allowed to restore an inoperable Alternate Shutdown System Function to OPERABLE status in the ITS than was allowed in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

TS

3.3 INSTRUMENTATION

3.13 3.3.3.2 [Remote] Shutdown System 1
Alternate

3.13.A.1 LCO 3.3.3.2 The [Remote] Shutdown System Functions shall be OPERABLE. 1

3.13.A.1 APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----
DOC A.4 Separate Condition entry is allowed for each Function.

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.13.A.2	A. One or more required Functions inoperable.	A.1 Restore required Function to OPERABLE status.	30 days
3.13.A.2.d	B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
DOC M.3	SR 3.3.3.2.1	[Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days]	2
4.13.A.1	SR 3.3.3.2.2	Verify each required control circuit and transfer switch is capable of performing the intended function.	[18] months 24	2
DOC M.3	SR 3.3.3.2.3	Perform CHANNEL CALIBRATION for each required instrumentation channel.	[18] months 24	2

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.3.2, ALTERNATE SHUTDOWN SYSTEM

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. The brackets have been removed and the proper plant specific information/value has been provided.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

 1

B 3.3 INSTRUMENTATION

B 3.3.3.2 ~~Remote~~ Alternate Shutdown System 1

BASES

BACKGROUND

The ~~Remote~~ Alternate Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the plant in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility of the control room becoming inaccessible. A safe shutdown condition is defined as MODE 3. With the plant in MODE 3, the ~~Reactor Core Isolation Cooling (RCIC)~~ Core Spray (CS) System, the safety/relief valves (S/RVs), and the Residual Heat Removal (RHR) ~~Shutdown Cooling~~ System operating in the suppression pool cooling mode can be used to remove core decay heat and meet all safety requirements. INSERT 1 The long term supply of water for the ~~RCIC~~ CS System and the ability to operate ~~shutdown cooling~~ the RHR System in the suppression pool cooling mode from outside the control room allow extended operation in MODE 3. 2

In the event that the control room becomes inaccessible, the operators can establish control at the ~~remote~~ Alternate shutdown panel INSERT 2 and place and System maintain the plant in MODE 3. Not all controls and necessary transfer switches are located at the remote shutdown panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The plant automatically reaches MODE 3 following a plant shutdown and can be maintained safely in MODE 3 for an extended period of time. 2 3

The OPERABILITY of the ~~Remote~~ Alternate Shutdown System control and instrumentation Functions ensures that there is sufficient information available on selected plant parameters to place and maintain the plant in MODE 3 should the control room become inaccessible. 2

APPLICABLE SAFETY ANALYSES

The ~~Remote~~ Alternate Shutdown System is required to provide equipment at appropriate locations outside the control room with a design capability to promptly shut down the reactor to MODE 3, including the necessary instrumentation and controls, to maintain the plant in a safe condition in MODE 3. 2

The criteria governing the design and the specific system requirements of the ~~Remote~~ Alternate Shutdown System are located in ~~10 CFR 50, Appendix A, GDC 19 (Ref. 1)~~ USAR, Sections 7.11.1 and 10.3.1.5.4 (Refs. 1 and 2). 2

The ~~Remote~~ Alternate Shutdown System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). 2

2

INSERT 1

A minimum of two S/RVs will be manually controlled at the Alternate Shutdown System panel to reduce Reactor Coolant System pressure. After depressurization, the CS System will provide reactor inventory makeup. The CS System will be used to establish a cooling path by allowing the reactor vessel water level to rise until water flows through the S/RV lines into the suppression pool. Decay heat removal is provided by manual operation of the RHR System in the suppression pool cooling mode.

2

INSERT 2

The design of the Alternate Shutdown System panel includes a master transfer switch which, when activated, enables Alternate Shutdown System operation, initiates an annunciator in the control room, and initiates an indication light and activates other transfer switches at the Alternate Shutdown System panel. It also includes a main steam isolation valve (MSIV) isolation switch and four system transfer switches which, when activated, will ensure closure of MSIVs and enable the manual control and operation of the four S/RVs, CS System, RHR System, and other auxiliary systems from the Alternate Shutdown System panel.

 1

BASES

LCO

Alternate

The ~~Remote~~ Shutdown System LCO provides the requirements for the OPERABILITY of the instrumentation and controls necessary to place and maintain the plant in MODE 3 from a location other than the control room. The instrumentation and controls required are from Division 2 and are listed in Table B 3.3.3.2-1. The Alternate Shutdown System panel transfer switch is also required to be OPERABLE. 2

The controls, instrumentation, and transfer switches are those required for:

- Reactor pressure vessel (RPV) pressure control ~~,~~ ; 3
- Decay heat removal ~~,~~ ; 3
- RPV inventory control ~~,~~ ; and 3
- Safety support systems for the above functions, including ~~service water, component cooling water, and onsite power, including the diesel generators~~ the RHR Service Water System and ECCS room coolers. 2

Alternate

The ~~Remote~~ Shutdown System is OPERABLE if all ~~instrument and control channels~~ System needed to support the ~~remote~~ Alternate ~~shutdown function~~ are OPERABLE. ~~In some cases, Table B 3.3.3.2-1 may indicate that the required information or control capability is available from several alternate sources. In these cases, the Remote Shutdown System is OPERABLE as long as one channel of any of the alternate information or control sources for each Function is OPERABLE.~~ 2

Alternate

The ~~Remote~~ Shutdown System instruments and control circuits covered by this LCO do not need to be energized to be considered OPERABLE. In addition, the Alternate Shutdown System master transfer switch is required to be locked in the normal position when the panel is not in use. This LCO is intended to ensure that the instruments and control circuits will be OPERABLE if plant conditions require that the ~~Remote~~ Alternate Shutdown System be placed in operation. 2

APPLICABILITY

Alternate

The ~~Remote~~ Shutdown System LCO is applicable in MODES 1 and 2. This is required so that the plant can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room. 2

This LCO is not applicable in MODES 3, 4, and 5. In these MODES, the plant is already subcritical and in a condition of reduced Reactor Coolant System energy. Under these conditions, considerable time is available to restore necessary instrument control Functions if control room instruments or control becomes unavailable. Consequently, the TS do not require OPERABILITY in MODES 3, 4, and 5.

BASES

ACTIONS A Remote Shutdown System division is inoperable when each function is not accomplished by at least one designated Remote Shutdown System channel that satisfies the OPERABILITY criteria for the channel's Function. These criteria are outlined in the LCO section of the Bases. (4) (2)

A Note has been provided to modify the ACTIONS related to Alternate Shutdown System Functions. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable Alternate Shutdown System Functions provide appropriate compensatory measures for separate Functions. As such, a Note has been provided that allows separate Condition entry for each inoperable Alternate Shutdown System Function. (2) (2) (2)

A.1

Condition A addresses the situation where one or more required Functions of the Alternate Shutdown System is inoperable. This includes the control and transfer switches for any required Function. (2)

The Required Action is to restore the Function ~~(both divisions, if applicable)~~ to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room. (2)

B.1

If the Required Action and associated Completion Time of Condition A are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.2.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. As specified in the Surveillance, a CHANNEL CHECK is only required for those channels that are normally energized.

The Frequency is based upon plant operating experience that demonstrates channel failure is rare.

SR 3.3.3.2.2

Alternate

SR 3.3.3.2.2 verifies each required Remote Shutdown System transfer switch and control circuit performs the intended function. This verification is performed from the remote shutdown panel and locally, as appropriate. (2)

In addition, for the master transfer switch, this SR ensures the alarm in the control room functions when the switch is in the transfer position.

Operation of the equipment from the remote shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the plant can be placed and maintained in MODE 3 from the (2)

alternate

remote shutdown panel and the local control stations. ~~However, this Surveillance is not required to be performed only during a plant outage.~~ (2) (7)

Alternate

Operating experience demonstrates that Remote Shutdown System control channels usually pass the Surveillance when performed at the (2)

24

18 month Frequency. (1)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.2.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies the channel responds to measured parameter values with the necessary range and accuracy.

The ~~18~~ 24 month Frequency is based upon operating experience and consistency with the typical industry refueling cycle. 1

REFERENCES 1. ~~10 CFR 50, Appendix A, GDC 19~~ USAR, Section 7.11.1 2

2. USAR, Section 10.3.1.5.4. 2

Alternate

Table B 3.3.3.2-1 (page 1 of 1)
Remote Shutdown System Instrumentation 1

Alternate

REQUIRED CHANNELS 5

FUNCTION (INSTRUMENT OR CONTROL PARAMETER)	~~REQUIRED NUMBER OF DIVISIONS~~
1. Reactor Pressure Vessel Pressure	
a. Reactor Pressure	[1]
INSERT 3	
2. Decay Heat Removal	
INSERT 4	
~~a. RCIC Flow~~	~~[1]~~
~~b. RCIC Controls~~	~~[1]~~
~~c. RHR Flow~~	~~[1]~~
~~d. RHR Controls~~	~~[1]~~
3. Reactor Pressure Vessel Inventory Control	
INSERT 5	
~~a. RCIC Flow~~	~~[1]~~
~~b. RCIC Controls~~	~~[1]~~
~~c. RHR Flow~~	~~[1]~~
~~d. RHR Controls~~	~~[1]~~

5 5 5 5

~~------REVIEWER'S NOTE------~~
~~For channels that fulfill GDC 19 requirements, the number of OPERABLE channels required depends upon the plant's licensing basis as described in the NRC plant specific Safety Evaluation Report (SER). Generally, two divisions are required to be OPERABLE. However, only one channel per given Function is required if the plant has justified such a design and the NRC SER has accepted the justification.~~
~~------------~~

~~This Table is for illustration purposes only. It does not attempt to encompass every Function used at every unit, but does contain the types of Functions commonly found.~~ 6

5

INSERT 3

b.	Safety/Relief Valve Transfer Switch	1
c.	Safety/Relief Valve Controls	2

5

INSERT 4

a.	RHR System Transfer Switch	1
b.	RHR Suppression Pool Cooling Flow	1
c.	RHR Suppression Pool Cooling Controls (includes RHR Service Water controls)	1
d.	RHR Service Water Flow	1
e.	Suppression Pool Water Level	1
f.	Suppression Pool Water Temperature (Average and Local)	1
g.	ECCS Room Cooler Controls	1

5

INSERT 5

a.	Reactor Vessel Water Level (Flooding Range)	1
b.	Reactor Vessel Water Level (Wide Range)	1
c.	Core Spray System Transfer Switch	1
d.	Core Spray Flow	1
e.	Core Spray Controls	1
f.	Main Steam Isolation Valve Isolation Switch	1

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.3.2 BASES, ALTERNATE SHUTDOWN SYSTEM

1. Changes have been made to reflect those changes made to the Specification.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. These punctuation corrections have been made consistent with the Writers Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

4. Typographical/grammatical error corrected.

5. The brackets have been removed and the proper plant specific information/value has been provided.

6. This Reviewer's Note (or reviewer's type of note) has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

7. Changes have been made to reflect the Specification.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.3.2, ALTERNATE SHUTDOWN SYSTEM

There are no specific NSHC discussions for this Specification.

ATTACHMENT 7

ITS 3.3.4.1, Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3.2 PROTECTIVE INSTRUMENTATION	**4.2 PROTECTIVE INSTRUMENTATION**
Applicability: ~~Applies to the plant instrumentation which performs a protective function.~~ Objective: ~~To assure the operability of protective instrumentation.~~ Specification:	Applicability: ~~Applies to the surveillance requirements of the instrumentation that performs a protective function.~~ Objective: ~~To specify the type and frequency of surveillance to be applied to protective instrumentation.~~
A. Primary Containment Isolation Functions When primary containment integrity is required, the limiting conditions of operation for the instrumentation that initiates primary containment isolation are given in Table 3.2.1.	Specification: ~~The instrumentation to be functionally tested and calibrated and the frequency of the tests is given in Table 4.2.1.~~

See ITS 3.3.6.1

A.1

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A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
F. Recirculation Pump Trip ~~and Alternate Rod Injection~~ Initiation	LA.1
Applicability / LCO 3.3.4.1 — Whenever the reactor is in the RUN mode, the Limiting Conditions for Operation for the instrumentation listed in Table 3.2.5 shall be met.	
G. Safeguards Bus Voltage Protection Whenever the safeguards auxiliary electrical power system is required to be operable by Specification 3.9, the Limiting Conditions for Operation for the instrumentation listed in Table 3.2.6 shall be met.	See ITS 3.3.8.1
H. Instrumentation for Safety/Relief Valve Low-Low Set Logic	See ITS 3.3.6.3 and ITS 3.6.1.5
Whenever the safety/relief valves are required to be operable by Specification 3.6.E, the Limiting Conditions for Operation for the instrumentation listed in Table 3.2.7 shall be met.	See ITS 3.3.6.3
I. Instrumentation for Control Room Habitability Protection 1. Whenever the emergency filtration system is required to be operable by Specification 3.17.B, the Limiting Conditions for Operation for the radiation instrumentation listed in Table 3.2.9 shall be met.	See ITS 3.3.7.1

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A.1

ITS

Table 3.3.4.1-1

Table 3.2.5
Instrumentation That Initiates a Recirculation Pump Trip and Alternate Rod Injection

ITS	Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems (1)	Total No. of Instrument Channels per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Conditions*
LCO 3.3.4.1.b, SR 3.3.4.1.5.b	1. High Reactor Dome Pressure	≤ 1150 psig	2	2	2	A
LCO 3.3.4.1.a, SR 3.3.4.1.5.a	2. Low-Low Reactor Water Level	≥ -48"	2	2	2	A

Allowable Value

Add time delay Allowable Value

1155

NOTE: Add proposed ACTIONS Note

ACTIONS A, B, C, and D 1. When one of the two trip systems is made or found to be inoperable, restore the inoperable trip system to operable status within 14 days or place the plant in the specified required condition within the next eight hours. When both trip systems are inoperable, place the plant in the specified required condition within eight hours unless at least one trip system is sooner made operable.

Add proposed ACTIONS A, B, and C

6

* Required conditions when minimum conditions for operation are not satisfied:

Required Action D.2 A. Reactor in Startup, Refuel, or Shutdown Mode

MODE 2

Add proposed Required Action D.1

M.1

A.6

LA.1

L.4

LA.2

L.1

M.2

A.2

A.2

L.2

A.1

ITS

Table 3.3.4.1-1 ← Add proposed Surveillance Requirements Note — L.3

Table 4.2.1 Continued
Minimum Test and Calibration Frequency for Core Cooling, Rod Block and Isolation Instrumentation — A.4

Instrument Channel	Test (2) SR 3.3.4.1.2	Calibration (2) SR 3.3.4.1.3, SR 3.3.4.1.5	Sensor Check (2) SR 3.3.4.1.1
REACTOR BUILDING VENTILATION & STANDBY GAS TREATMENT			
1. Reactor Low Low Water Level	Once/3 months (Note 5)	Every Operating Cycle - Transmitter Once/3 months - Trip Unit	Once/12 hours
2. Drywell High Pressure (Note 10)	*	*	*
3. Radiation Monitors (Plenum)	Once/3 months	Once/3 months	Once/day
4. Radiation Monitors (Refueling Floor)	Once/3 months	Once/3 months	Note 4
RECIRCULATION PUMP TRIP AND ALTERNATE ROD INJECTION			
LCO 3.3.4.1.b 1. Reactor High Pressure	Once/3 months (Note 5) -2	Once/Operating Cycle 24 months Transmitter-5 Once/3 Months-Trip Unit -3	Once/Day 12 hours -1
LCO 3.3.4.1.a 2. Reactor Low Low Water Level	Once/3 months (Note 5) -2	Once/Operating Cycle - Transmitter-5 Once/3 Months-Trip Unit -3	Once/12 hours-1
SHUTDOWN COOLING SUPPLY ISOLATION			
1. Reactor Pressure Interlock	Once/3 months	Once/3 Months	None
SAFEGUARDS BUS VOLTAGE			
1. Degraded Voltage Protection	Once/month	Quarterly	Not applicable
2. Loss of Voltage Protection	Once/month	Once/Operating Cycle	Not applicable
SAFETY/RELIEF VALVE LOW-LOW SET LOGIC			
1. Reactor Scram Sensing	Once/Shutdown (Note 8)	*	*
2. Reactor Pressure - Opening	Once/3 months (Note 5)	Once/Operating Cycle	Once/day
3. Reactor Pressure - Closing	Once/3 months (Note 5)	Once/Operating Cycle	Once/day
4. Discharge Pipe Pressure	Once/3 months (Note 5)	See Table 4.14.1	See Table 4.14.1
5. Inhibit Timer	Once/3 months (Note 5)	Once/Operating Cycle	*
CONTROL ROOM HABITABILITY PROTECTION			
1. Radiation	Monthly (Note 5)	18 months	Daily

See ITS 3.3.6.2

LA.1

M.3

A.5

A.3

See ITS 3.3.6.1

See ITS 3.3.8.1

See ITS 3.3.6.3

See ITS 3.3.7.1

3.2/4.2 63 03/07/01
Amendment No. 62, 63, 65, 89, 117

Add proposed SR 3.3.4.1.6 — M.4

Add proposed SR 3.3.4.1.4 — M.1

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A.1

Table 4.2.1 Continued
Minimum Test and Calibration Frequency for Core Cooling,
Rod Block and Isolation Instrumentation

NOTES:

(1) (Deleted)

(2) Calibrate prior to normal shutdown and start-up and thereafter check once per 12 hours and test once per week until no longer required. Calibration of this instrument prior to normal shutdown means adjustment of channel trips so that they correspond, within acceptable range and accuracy, to a simulated signal injected into the instrument (not primary sensor). In addition, IRM gain adjustment will be performed, as necessary, in the APRM/IRM overlap region. See ITS 3.3.2.1

(3) Functional tests, calibrations and sensor checks are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status. A.4

(4) Whenever fuel handling is in process, a sensor check shall be performed once per 12 hours. See ITS 3.3.6.2

(5) A functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and/or initiating action. A.5

(6) (Deleted)

(7) (Deleted)

(8) Once/shutdown if not tested during previous 3 month period. See ITS 3.3.6.3

(9) Testing of the SRM Not-Full-In rod block is not required if the SRM detectors are secured in the full-in position. See ITS 3.3.2.1

(10) Uses contacts from scram system. Tested and calibrated in accordance with Tables 4.1.1 and 4.1.2. See ITS 3.3.6.1 and ITS 3.3.6.2

(11) Uses contacts from Group 1 isolation logic. Tested and calibrated in accordance with Group 1 Low Low Water Level Instrumentation.

(12) Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. See ITS 3.3.6.1

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DISCUSSION OF CHANGES
ITS 3.3.4.1, ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP (ATWS-RPT) INSTRUMENTATION

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS Table 3.2.5 Required Condition A requires the plant to be in Startup, Refuel, or Shutdown Mode if the Required Actions provided in Note 1 are not met. When the Required Actions and associated Conditions are not met in the ITS, ITS 3.3.4.1 Required Action D.2 requires the plant to be in MODE 2. This changes the CTS by only specifying a default action to be in MODE 2 (Startup) instead of providing the option to be in the Refuel or Shutdown Mode.

The purpose of the CTS Table 3.2.5 Required Condition A is to place the plant in a condition where ATWS-RPT Functions are not required to be OPERABLE. ITS 3.3.4.1 Required Action D.2 requires the plant to be in MODE 2. ITS Table 1.1-1 defines MODE 2 when the reactor mode switch is in the Startup/Hot Standby or Refuel position with the head on the vessel. CTS requires the reactor to be placed in any mode other than Run. This change is acceptable because the proposed Required Action still places the unit outside the Applicability of the Specification. The ATWS-RPT functions are not required to mitigate the consequences of an ATWS event when the reactor mode switch is in the Startup/Hot Standby, Refuel position, or Shutdown. This change is administrative because the reactor mode switch must still be placed in a position other than Run.

A.3 CTS Table 4.2.1 Instrument Channels 1 and 2 associated with the Recirculation Pump Trip Instrumentation specifies an "Operating Cycle" Frequency for the CHANNEL CALIBRATION of the transmitter. ITS SR 3.3.4.1.5 requires the performance of a CHANNEL CALIBRATION every "24 months." This changes the CTS by changing the Frequency from once per "Operating Cycle" to "24 months."

This change is acceptable because the current "Operating Cycle" is "24 months." In letter L-MT-04-036, from Thomas J. Palmisano (NMC) to the USNRC, dated June 30, 2004, NMC has proposed to extend the fuel cycle from 18 months to 24 months and at the same time has performed an evaluation in accordance with Generic Letter 91-04 to extend the unit Surveillance Requirements from 18 months to 24 months. CTS Table 4.2.1 was included in this evaluation. This change is designated as administrative because it does not result in any technical changes to the CTS.

A.4 CTS Table 4.2.1 Note (3) states that, functional tests, calibrations, and sensor checks are not required when the systems are not required to be OPERABLE or are tripped. In addition, the Note states that if tests are missed, they shall be performed prior to returning the systems to an OPERABLE status. These explicit requirements are not retained in ITS 3.3.4.1. This changes the CTS by not including these explicit requirements.

The purpose of this Note is to provide guidance on when Surveillances are required to be met and performed. This explicit Note is not needed in ITS 3.3.4.1 since these allowances are included in ITS SR 3.0.1. ITS SR 3.0.1 states that SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR, and failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. SR 3.0.1 also states that SRs are not required to be performed on inoperable equipment. When equipment is declared inoperable, the Actions of this LCO require the equipment to be placed in the trip condition. In this condition, the equipment is still inoperable but has accomplished the required safety function. Therefore the allowances in SR 3.0.1 and the associated actions provide adequate guidance with respect to when the associated surveillances are required to be performed and this explicit requirement is not retained. This change is administrative because it does not result in a technical change to the CTS.

A.5 CTS Table 4.2.1 Note (5) states that functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response, alarm, and/or initiating action. These explicit requirements are not retained in ITS 3.3.4.1. This changes the CTS by not including these explicit requirements.

The purpose of CTS Table 4.2.1 Note (5) is to provide guidance on how to perform an instrument functional test of the ATWS-RPT instrument channels. This explicit Note is not needed in ITS 3.3.4.1 since the requirements for the CHANNEL FUNCTIONAL TEST are included in ITS 1.0, "Definitions." ITS 1.0 states that a CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. Therefore, the ITS 1.0 definition provides adequate guidance with respect to performance requirements of a CHANNEL FUNCTIONAL TEST and this explicit requirement is not retained. This change is administrative because it does not result in a technical change to the CTS.

A.6 CTS 3.2.F states that the Limiting Conditions for Operation for the instrumentation listed in Table 3.2.5 shall be met. CTS Table 3.2.5 specifies the "Trip Setting" for each ATWS-RPT instrument Function. ITS LCO 3.3.4.1 requires the ATWS-RPT instrumentations for each Function to be OPERABLE and ITS SR 3.3.4.1.5 specifies the "Allowable Value" for each Function. This changes the CTS by replacing the term "Trip Setting" with "Allowable Value."

The purpose of the "Trip Setting" in CTS Table 3.2.5 is to define the OPERABILITY limits for the ATWS-RPT instrumentation Functions. Therefore, the use of the term "Trip Setting" in the CTS is the same as the use of the term "Allowable Value" in the ITS. This proposed change does not modify the actual "Trip Setting" specified in CTS Table 3.2.5 for the ATWS-RPT instrumentation

DISCUSSION OF CHANGES
ITS 3.3.4.1, ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP (ATWS-RPT) INSTRUMENTATION

Functions. Any changes to the actual value specified in the "Trip Setting" (i.e., changing the value for OPERABILITY) are discussed in DOC L.4. This change is designated as administrative change and is acceptable because it does not result in any technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS Table 3.2.5 provides the Trip Setting for the ATWS-RPT Low-Low Reactor Water Level Function. However, it does not specify the requirements for the time delay relays associated with each ATWS-RPT Low-Low Reactor Water Level channel. ITS SR 3.3.4.1.4 requires the Allowable Value for the time delay relay portion of the ATWS-RPT Reactor Vessel Water Level - Low Low channels to be $\geq$ 6.0 seconds and $\leq$ 8.6 seconds and adds a Surveillance Requirement to perform a CHANNEL CALIBRATION every 184 days. This changes the CTS by providing explicit values for the time delay relays associated with the ATWS-RPT Low-Low Reactor Water Level channels and adding a Surveillance Requirement to perform a CHANNEL CALIBRATION every 184 days.

This change is acceptable because proper settings and testing of the time delay relays of the ATWS-RPT Reactor Vessel Water Level - Low Low channels are necessary to support the OPERABILITY of the ATWS-RPT Reactor Vessel Water Level - Low Low Function. As such, explicitly including the values for the time delay relays and testing requirements in the Technical Specifications provides additional assurance that the OPERABILITY of the ATWS-RPT Reactor Vessel Water Level - Low Low channels will be maintained. The addition of the time delay relay Allowable Values of the ATWS-RPT Reactor Vessel Water Level - Low Low channels is acceptable since these requirements and the testing requirements are currently administratively controlled in procedures. The requirements for the ATWS-RPT Reactor Vessel Water Level - Low Low Function continues to require the time delay relay Allowable Values to be within required limits to ensure that these instruments function as assumed in the safety analyses. This change is designated as more restrictive because it adds explicit Allowable Values for the ATWS-RPT Reactor Vessel Water Level - Low Low Function.

M.2 CTS Table 3.2.5 Note 1 requires, when a trip system of one Function is inoperable and not restored within 14 days or if both trip systems of a Function are inoperable, the plant to be in a condition other than Run within 8 hours. Under the same conditions in the ITS, ITS 3.3.4.1 Required Action D.2 requires the plant to be in MODE 2 within 6 hours. This changes the CTS by reducing the time the plant must be in MODE 2 from 8 hours to 6 hours.

The purpose of the shutdown actions of CTS Table 3.2.5 Note 1 is to place the plant outside of the Applicability of the Specification. ITS 3.3.4.1 Required Action D.2 continues to accomplish this purpose, but the time to be in MODE 2 has decreased from 8 hours to 6 hours. This change is acceptable because the time required to be in MODE 2 is reasonable, based on operating experience, both to reach MODE 2 from full power conditions and to remove a recirculation pump from service in an orderly manner and without challenging plant systems.

DISCUSSION OF CHANGES
ITS 3.3.4.1, ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP (ATWS-RPT) INSTRUMENTATION

This change is designated as more restrictive because it reduces the amount of time provided to complete a Required Action.

M.3 CTS Table 4.2.1 requires the performance of a sensor check on the ATWS-RPT Reactor High Pressure channels "Once/Day." ITS SR 3.3.4.1.1 requires the performance of a CHANNEL CHECK every 12 hours. This changes the CTS by increasing the Surveillance Frequency from "Once/Day" to every "12 hours."

The purpose of the CTS Table 4.2.1 sensor checks is to ensure the channels are within plant agreement criteria. This change is acceptable because it helps to ensure the Function is maintained OPERABLE. This change is consistent with BWR ISTS, NUREG-1433, Rev. 3, and the current requirements for other instrumentation (i.e., Reactor Vessel Water Level - Low Low for ATWS-RPT) within the CTS. This change is designated as more restrictive because the ITS will require the Surveillance to be performed more frequently than in the CTS.

M.4 CTS Table 4.2.1 does not specify requirements for a LOGIC SYSTEM FUNCTIONAL TEST. ITS 3.3.4.1 requires the performance of SR 3.3.4.1.6, a LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation every 24 months, for each ATWS-RPT Function. This changes the CTS by explicitly requiring a LOGIC SYSTEM FUNCTIONAL TEST, including breaker actuation to be performed on each ATWS-RPT Function.

This change is acceptable because a LOGIC SYSTEM FUNCTIONAL TEST helps to ensure the ATWS-RPT logic is functioning as required to support the safety analyses. As such, explicitly including requirements for a LOGIC SYSTEM FUNCTIONAL TEST in the Technical Specifications provides additional assurance that the OPERABILITY of the ATWS-RPT Instrumentation Functions will be maintained. This change is designated as more restrictive because it adds a specific requirement to perform a LOGIC SYSTEM FUNCTIONAL TEST on each ATWS-RPT Instrumentation Function.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 6 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAPD, or IIP)* CTS 3.2.F, CTS Table 3.2.5, and CTS Table 4.2.1 provide requirements for the Alternate Rod Injection Instrumentation. ITS 3.3.4.1 does not include requirements for the Alternate Rod Injection Instrumentation. This changes the CTS by moving the explicit Alternate Rod Injection Instrumentation requirements from the Technical Specifications to the Technical Requirements Manual (TRM). The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to provide adequate protection of public health and safety.

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ITS 3.3.4.1, ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP (ATWS-RPT) INSTRUMENTATION

The purpose of CTS 3.2.F, CTS Table 3.2.5, and the associated requirements in CTS Table 4.2.1, in part, is to ensure that the initiation feature of the Alternate Rod Injection is OPERABLE. NUREG-1433, Rev. 3 does not provide requirements for the Alternate Rod Injection Instrumentation, therefore, the CTS requirements have been relocated to the TRM. The Alternate Rod Injection Instrumentation capability is still required to be maintained because Monticello is still required to satisfy the ATWS design requirements of 10 CFR 50.62, the ATWS Rule. The Alternate Rod Injection Instrumentation is part of these requirements. This change is acceptable because the removed LCO, Applicability, Actions, and Surveillance Requirements will be adequately controlled in the TRM. The TRM is incorporated by reference into the USAR and any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because a requirement is being removed from the Technical Specifications.

LA.2 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.2.5 specifies the "Minimum No. of Operable or Operating Trip Systems" and "Total No. of Instrument Channels Per Trip System" for the ATWS-RPT Functions. ITS 3.3.4.1 does not include these details. This changes the CTS by moving the information of the "Minimum No. of Operable or Operating Trip Systems" and "Total No. of Instrument Channels Per Trip System" to the ITS Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the minimum number of required channels per trip system for each ATWS-RPT instrumentation Function. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 3 – Relaxation of Completion Time)* CTS Table 3.2.5 Note 1 provides an Action for an inoperable ATWS-RPT trip system (i.e., one or two channels in the trip system inoperable) and allows 14 days to restore the trip system to OPERABLE status. If the trip system is not restored to OPERABLE status, or if both trip systems are inoperable, the plant must be placed in at least Startup in 8 hours. ITS 3.3.4.1 includes an ACTIONS Note that allows separate Condition entry for each channel. ITS 3.3.4.1 ACTION A covers the condition for one or more channels inoperable, and allows either 14 days to restore the channel to OPERABLE status or to place the channel in trip. The allowance to place the channel in trip is not applicable if the inoperable channel is the result of an

DISCUSSION OF CHANGES
ITS 3.3.4.1, ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP (ATWS-RPT) INSTRUMENTATION

inoperable breaker. ITS 3.3.4.1 ACTION B covers the condition of one Function (Reactor Vessel Water Level - Low Low or Reactor Vessel Steam Dome Pressure - High) with ATWS-RPT trip capability not maintained (i.e., both trip systems for a Function inoperable), and requires restoration of ATWS-RPT trip capability (i.e., restoration of one of the two trip systems) within 72 hours. If both ATWS-RPT Functions do not have trip capability (i.e., both trip systems for a Function inoperable), ITS 3.3.4.1 ACTION C requires restoration of the ATWS-RPT trip capability (i.e., restoration of one of the two trip systems) for one Function within 1 hour. This changes the CTS in several ways: a) it allows 14 days to restore each inoperable channel instead of the current requirement to restore all channels in a trip system to OPERABLE status in 14 days; b) it allows an inoperable channel to be placed in trip in lieu of restoring the channel to OPERABLE status; c) it allows 72 hours to restore ATWS-RPT trip capability (i.e., restore one of the two trip systems) for a Function that has two inoperable trip systems prior to requiring a plant shutdown to MODE 2; and d) when both Functions have two inoperable trip systems, it allows 1 hour to restore ATWS-RPT trip capability (i.e., restore one of the two trip systems) for one of the two Functions prior to requiring a plant shutdown to MODE 2.

The purpose of CTS Table 3.2.5 Note 1 is to allow only a short time (14 days) to restore one inoperable trip system associated with an ATWS-RPT trip Function before commencing a reactor shutdown. This change is acceptable because the Completion Time is consistent with the safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of an ATWS event occurring during the allowed Completion Time. When a single trip system for a Function is inoperable under the CTS requirements, either due to one or two inoperable reactor vessel water level channels or one or two inoperable reactor vessel pressure channels, or both, the ITS will not have an inoperable Function. This is acceptable because while in this condition, the ATWS-RPT System is still capable of tripping both recirculation pumps on either Function. In this case, ITS 3.3.4.1 ACTION A applies to each inoperable channel and 14 days is allowed to either restore each inoperable channel to OPERABLE status or to place the channel in trip. This allowed out of service time has been shown to maintain an acceptable risk in accordance with previously conducted reliability analysis (GENE-770-06-1-A, December 1992). The logic design of ATWS-RPT instrumentation is bounded by this reliability analysis and the conclusions of the analysis are applicable to the Monticello design. The result of the NRC review of this generic reliability analyses as it relates to Monticello is documented in the NRC Safety Evaluation Report (SER) dated December 23, 1998. The SER concluded that the generic reliability analysis is applicable to Monticello, and that Monticello meets all requirements of the NRC SER accepting the generic reliability analysis. When both trip systems are inoperable for a Function under the CTS requirements due to one or two channels of the same Function being inoperable in both trip systems, the plant must be placed in at least MODE 2. In the ITS, when two channels of the same Function are inoperable in both trip systems, one Function will be inoperable. Therefore, ITS 3.3.4.1 ACTION B would apply, and allows 72 hours to restore the ATWS-RPT trip capability. This is acceptable since while in this condition, the

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ITS 3.3.4.1, ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP (ATWS-RPT) INSTRUMENTATION

ATWS-RPT System is still capable of tripping both recirculation pumps on the other Function and operator action can still be taken to trip the recirculation pumps during this beyond design basis event. Trip capability is maintained for any Function as long as there are two OPERABLE or tripped channels in the same trip system. When both trip systems for both Functions are inoperable under the CTS requirements due to one or two channels of both Functions being inoperable in both trip systems, the plant must be placed in at least MODE 2. In the ITS, under the same conditions, both Functions are considered inoperable. Therefore, ITS 3.3.4.1 ACTION C would apply, and allows one hour to restore ATWS-RPT trip capability. The 1 hour Completion Time is acceptable because it provides sufficient time for the operator to take corrective action and takes into account the likelihood of an event requiring actuation of the ATWS-RPT instrumentation during this period. This change is less restrictive because more time is allowed in the ITS to restore or place the channels in trip than is allowed in the CTS.

L.2 *(Category 4 – Relaxation of Required Action)* CTS Table 3.2.5 Required Condition A requires the unit to be in Startup, Refuel or Shutdown Mode if the Required Actions provided in Note 1 are not met. ITS 3.3.4.1 Required Action D.2 includes a similar requirement, but ITS 3.3.4.1 Required Action D.1 also allows the removal of the affected recirculation pump from service in lieu of shutdown to MODE 2. This Required Action is only applicable if the inoperable channel is the result of an inoperable breaker. This changes the CTS by allowing the breaker to be tripped instead of exiting the MODE 1.

The purpose of CTS Table 3.2.5 Required Condition A is to place the plant in a condition where the instruments are not required. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. The purpose of the ATWS-RPT instrumentation is to trip the recirculation pumps. Therefore, an additional Required Action is proposed, ITS 3.3.4.2 Required Action D.1, to allow removal of the associated recirculation pump from service in lieu of being in MODE 2. Since this action accomplishes the functional purpose of the ATWS-RPT instrumentation and enables continued operation in a previously approved condition, this change does not have a significant effect on safe operation. This change is less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.3 *(Category 4 – Relaxation of Required Action)* CTS Table 3.2.5 does not provide a delayed entry into associated Conditions and Required Actions if an ATWS-RPT channel is inoperable solely for performance of required Surveillances. The ITS 3.3.4.1 Surveillance Requirements Note allows delayed entry into associated Conditions and Required Actions for up to 6 hours if an ATWS-RPT channel is placed in an inoperable status solely for performance of required Surveillances, provided the associated Function maintains ATWS-RPT trip capability. This changes the CTS by providing a delay time to enter Conditions and Required Actions for an ATW-RPT channel placed in an inoperable status solely for performance of required Surveillances.

The purpose of the ITS 3.3.4.1 Surveillance Requirements Note is to allow delayed entry into associated Conditions and Required Actions for up to 6 hours if an ATWS-RPT trip channel is placed in an inoperable status solely for performance of required Surveillances, provided the associated Function maintains ATWS-RPT trip capability. This change is acceptable because it is based on the reliability analysis assumption in GENE-770-06-1-A, December 1992. The result of the NRC review of this generic reliability analyses as it relates to Monticello is documented in the NRC Safety Evaluation Report (SER) dated December 23, 1998. The SER concluded that the generic reliability analysis is applicable to Monticello, and that Monticello meets all requirements of the NRC SER accepting the generic reliability analysis. This change is less restrictive because less stringent Required Actions are being applied in the ITS than were applied in CTS.

L.4 *(Category 10 – Changing Instrumentation Allowable Values)* CTS Table 3.2.5 specifies the "Trip Setting" for the ATWS-RPT High Reactor Dome Pressure Function. The Trip Setting value of CTS Table 3.2.5 Function 1 has been modified to reflect the new Allowable Value as indicated in ITS SR 3.3.4.1.5.b. This changes the CTS by requiring the ATWS-RPT instrumentation to be set consistent with the new "Allowable Value." The change in the term "Trip Setting" to "Allowable Value" is discussed in DOC A.6.

The purpose of the Allowable Values is to ensure the instruments function as assumed in the safety analyses. ITS SR 3.3.4.1.5.b reflects an Allowable Value consistent with the philosophy of General Electric ISTS, NUREG-1433. This Allowable Value has been established using the GE setpoint methodology guidance, as specified in the Monticello setpoint methodology. The analytic limits are derived from limiting values of the process parameters obtained from the safety analysis. The Allowable Value is derived from the analytic limits. The difference between the analytic limit and the Allowable Value allows for channel instrument accuracy, calibration accuracy, process measurement accuracy, and primary element accuracy. The margin between the Allowable Value and the NTSP allows for instrument drift that might occur during the established surveillance period. Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift. These two verifications are statistical evaluations to provide additional assurance of the acceptability of the NTSP and may require changes to the NTSP. Use of these methods and verifications provides the assurance that if the setpoint is found conservative to the Allowable Value during surveillance testing, the instrumentation would have provided the required trip function by the time the process reached the analytic limit for the applicable events. Therefore, based on the above discussion, the inclusion of the Allowable Value as the OPERABILITY value in lieu of the Trip Setting is

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acceptable. This change is designated as less restrictive because less stringent OPERABILITY values are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.3 INSTRUMENTATION

3.3.4.2 1 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation 1

3.2.F LCO 3.3.4.2 1 Two channels per trip system for each ATWS-RPT instrumentation Function listed below shall be OPERABLE: 1

Table 3.2.5 Function 2 a. Reactor Vessel Water Level - Low Low ~~Level 2~~ ; and 2 3

Table 3.2.5 Function 1 b. Reactor Vessel Steam Dome Pressure - High. 3

APPLICABILITY: MODE 1.

ACTIONS

-----NOTE-----
DOC L.2 Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.2.5 Note 1 A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status. <u>OR</u> A.2 -----NOTE----- Not applicable if inoperable channel is the result of an inoperable breaker. ----- Place channel in trip.	14 days 14 days 4
Table 3.2.5 Note 1 B. One Function with ATWS-RPT trip capability not maintained.	B.1 Restore ATWS-RPT trip capability.	72 hours

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME	
Table 3.2.5 Note 1	C. Both Functions with ATWS-RPT trip capability not maintained.	C.1 Restore ATWS-RPT trip capability for one Function.	1 hour	
Table 3.2.5 Note 1	D. Required Action and associated Completion Time not met.	D.1 ------------NOTE------------ Only applicable if inoperable channel is the result of an inoperable RPT breaker. ------------------------------		3
		Remove the affected recirculation pump from service.	6 hours	
		OR		
Table 3.2.5 Note 1, Table 3.2.5 Condition A		D.2 Be in MODE 2.	6 hours	

SURVEILLANCE REQUIREMENTS

--NOTE--

DOC L.4 When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability.

CTS	SURVEILLANCE		FREQUENCY	
Table 4.2.1 RPT Instruments 1 and 2	SR 3.3.4.2.1 1	[Perform CHANNEL CHECK. ——NOTE—— Not required for the time delay portion of the Reactor Vessel Water Level - Low Low Function.	12 hours]	1 5 6
Table 4.2.1 RPT Instruments 1 and 2	SR 3.3.4.2.2 1	Perform CHANNEL FUNCTIONAL TEST.	[92] days	1 5
Table 4.2.1 RPT Instruments 1 and 2	SR 3.3.4.2.3 1	[Calibrate the trip units.	[92] days]	1 5

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY	
	INSERT 1		6
Table 4.2.1 RPT Instruments 1 and 2	SR 3.3.4.~~2~~.~~4~~ 1 5 Perform CHANNEL CALIBRATION. The Allowable Values shall be:	24 ~~[18]~~ months	1 5 6
Table 3.2.5 Function 2, Table 4.2.1 RPT Instrument 2	a. Reactor Vessel Water Level - Low Low, ~~Level 2:~~ ≥ -48 ~~[-47]~~ inches, ~~and~~ ;		3 5 2
Table 3.2.5 Function 1, Table 4.2.1 RPT Instrument 1	b. Reactor Vessel Steam Dome Pressure - High ~~:~~ ≤ 1155 ~~[1095]~~ psig.		3 4 5
DOC M.3	SR 3.3.4.~~2~~.~~5~~ 1 6 Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	24 ~~[18]~~ months	1 5 6

CTS

3.3.4.1

6 **INSERT 1**

DOC M.1	SR 3.3.4.1.4	Perform CHANNEL CALIBRATION of Reactor Vessel Water Level - Low Low time delay relays. The Allowable Value shall be ≥ 6 seconds and ≤ 8.6 seconds.	184 days

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.4.1, ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP (ATWS-RPT) INSTRUMENTATION

1. ISTS 3.3.4.2 is renumbered as ITS 3.3.4.1 since ISTS 3.3.4.1, "End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation," is not included in the Monticello ITS.

2. These punctuation corrections have been made consistent with the Writer's Guide for the Improved Standard Technical Specifications, NEI 01-03, Section 5.1.3.

3. The proper Monticello plant specific nomenclature/value/design requirements have been provided.

4. Typographical error corrected.

5. The brackets have been removed and the proper specific information/value has been provided.

6. The Reactor Vessel Water Level - Low Low Function channels include a time delay relay in the channel circuitry for each level channel. ITS SR 3.3.4.1.4 has been added to require a CHANNEL CALIBRATION of the time delay relays to help ensure that the Reactor Vessel Water Level - Low Low Function channels provide a trip signal when necessary to satisfy the ATWS analysis. Subsequent SRs have been renumbered as necessary. In addition, a Note has been added to SR 3.3.4.1.1 (the CHANNEL CHECK) that states the Surveillance is not required for the time delay portion of the Reactor Vessel Water Level - Low Low Function since a CHANNEL CHECK cannot be performed on this portion of the channel (i.e., no indication is provided).

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION

B 3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation 1

BASES

BACKGROUND

The ATWS-RPT System initiates an RPT, adding negative reactivity, following events in which a scram does not (but should) occur, to lessen the effects of an ATWS event. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases. When Reactor Vessel Water Level - Low Low, Level 2 or Reactor Steam Dome Pressure - High setpoint is reached, the recirculation pump drive motor breakers trip.

Vessel

motor generator (MG) set

field

The ATWS-RPT System (Ref. 1) includes sensors, relays, bypass capability circuit breakers, and switches that are necessary to cause initiation of an RPT. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an ATWS-RPT signal to the trip logic.

Vessel Steam Dome

The ATWS-RPT consists of two independent trip systems, with two channels of Reactor Steam Dome Pressure - High and two channels of Reactor Vessel Water Level - Low Low, Level 2 in each trip system. Each ATWS-RPT trip system is a two-out-of-two logic for each Function. Thus, either two Reactor Water Level - Low Low, Level 2 or two Reactor Pressure - High signals are needed to trip a trip system. The outputs of the channels in a trip system are combined in a logic so that either trip system will trip both recirculation pumps (by tripping the respective drive motor breakers).

Vessel

Vessel

MG set

field

INSERT 1

There is one drive motor breaker provided for each of the two recirculation pumps for a total of two breakers. The output of each trip system is provided to both recirculation pump breakers.

MG set

MG set drive motor field

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ATWS-RPT is not assumed in the safety analysis. The ATWS-RPT initiates an RPT to aid in preserving the integrity of the fuel cladding following events in which a scram does not, but should, occur. ATWS-RPT instrumentation satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

to mitigate any accident or transient in the original design or licensing basis

The OPERABILITY of the ATWS-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.2.4. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Channel OPERABILITY also

or SR 3.3.4.1.5, as applicable

3 **INSERT 1**

Each Reactor Vessel Water Level - Low Low output must remain below the setpoint for approximately 7 seconds for the channel output to provide an actuation signal to the associated trip system.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

MG set field

includes the associated recirculation pump drive motor breakers. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. 3

Allowable Values are specified for each ATWS-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for. 2

ATWS (Ref. 2) 3

INSERT 2 2

The individual Functions are required to be OPERABLE in MODE 1 to protect against common mode failures of the Reactor Protection System by providing a diverse trip to mitigate the consequences of a postulated ATWS event. The Reactor Steam Dome Pressure - High and Reactor Vessel Water Level - Low Low, Level 2 Functions are required to be OPERABLE in MODE 1, since the reactor is producing significant power and the recirculation system could be at high flow. During this MODE, the potential exists for pressure increases or low water level, assuming an ATWS event. In MODE 2, the reactor is at low power and the recirculation system is at low flow; thus, the potential is low for a pressure increase or low water level, assuming an ATWS event. Therefore, the ATWS-RPT is not necessary. In MODES 3 and 4, the reactor is shut down with all control rods inserted; thus, an ATWS event is not significant and the possibility of a significant pressure increase or low water level is negligible. In MODE 5, the one rod out interlock ensures that the reactor remains subcritical; thus, an ATWS event is not significant. In addition, the reactor pressure vessel (RPV) head is not fully tensioned and no pressure transient threat to the reactor coolant pressure boundary (RCPB) exists.

Vessel 1 1

Instrumentation 2

3

INSERT 2

The Allowable Values and nominal trip setpoints (NTSP) are derived, using the General Electric setpoint methodology guidance, as specified in the Monticello setpoint methodology. The Allowable Values are derived from the analytic limits. The difference between the analytic limit and the Allowable Value allows for channel instrument accuracy, calibration accuracy, process measurement accuracy, and primary element accuracy. The margin between the Allowable Value and the NTSP allows for instrument drift that might occur during the established surveillance period. Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift. These two verifications are statistical evaluations to provide additional assurance of the acceptability of the NTSP and may require changes to the NTSP. Use of these methods and verifications provides the assurance that if the setpoint is found conservative to the Allowable Value during surveillance testing, the instrumentation would have provided the required trip function by the time the process reached the analytic limit for the applicable events.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The specific Applicable Safety Analyses and LCO discussions are listed below on a Function by Function basis.

a. Reactor Vessel Water Level - Low Low, ~~Level 2~~ 1

low low reactor vessel water level

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the ATWS-RPT System is initiated at ~~Level 2~~ to aid in maintaining level above the top of the active fuel. The reduction of core flow reduces the neutron flux and THERMAL POWER and, therefore, the rate of coolant boiloff. 1

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

Four channels of Reactor Vessel Water Level - Low Low, ~~Level 2,~~ with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. INSERT 3 The Reactor Vessel Water Level - Low Low, ~~Level 2~~ Allowable Value is chosen so that the system will not be initiated after a ~~Level 3~~ scram with feedwater still available, and for convenience with the reactor core isolation cooling initiation. INSERT 4 1 3 1 3 3 3

reactor vessel water level

Emergency Core Cooling System and

System

b. Reactor Steam Dome Pressure - High 1 3

Vessel

Excessively high RPV pressure may rupture the RCPB. An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This increases neutron flux and THERMAL POWER, which could potentially result in fuel failure and overpressurization. The Reactor Steam Dome Pressure - High Function initiates an RPT for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation. For the overpressurization event, the RPT aids in the termination of the ATWS event and, along with the safety/relief valves, limits the peak RPV pressure to less than the ASME Section III Code Service Level C limits (1500 psig). 1

Vessel

3 **INSERT 3**

Each channel includes a time delay relay that delays the Reactor Vessel Water Level - Low Low Function channel output signal from providing input to the associated trip system.

3 **INSERT 4**

The Reactor Vessel Water Level - Low Low Function trip is delayed since there is an insignificant effect on the ATWS consequences and it is desirable to avoid making the consequences of a loss of coolant accident more severe.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Vessel

The Reactor Steam Dome Pressure - High signals are initiated from four pressure transmitters that monitor reactor steam dome pressure. 1 Four channels of Reactor Steam Dome Pressure - High, with two channels in each trip system, are available and are required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Steam Dome Pressure - High Allowable Value is chosen to provide an adequate margin to the ASME Section III Code Service Level C allowable Reactor Coolant System pressure. 1 1 1

ACTIONS

A Note has been provided to modify the ACTIONS related to ATWS-RPT instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable ATWS-RPT instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable ATWS-RPT instrumentation channel.

A.1 and A.2

trip

With one or more channels inoperable, but with ATWS-RPT capability for each Function maintained (refer to Required Actions B.1 and C.1 Bases), the ATWS-RPT System is capable of performing the intended function. However, the reliability and redundancy of the ATWS-RPT instrumentation is reduced, such that a single failure in the remaining trip system could result in the inability of the ATWS-RPT System to perform the intended function. Therefore, only a limited time is allowed to restore the inoperable channels to OPERABLE status. Because of the diversity of sensors available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of ATWS-RPT, 14 days is provided to restore the inoperable channel (Required Action A.1). Alternately, the inoperable channel may be placed in trip (Required Action A.2), since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an 2

BASES

ACTIONS (continued)

inoperable breaker, since this may not adequately compensate for the inoperable breaker (e.g., the breaker may be inoperable such that it will not open). If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an RPT), or if the inoperable channel is the result of an inoperable breaker, Condition D must be entered and its Required Actions taken. 2

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining ATWS-RPT trip capability. A Function is considered to be maintaining ATWS-RPT trip capability when sufficient channels are OPERABLE or in trip such that the ATWS-RPT System will generate a trip signal from the given Function on a valid signal, and both recirculation pumps can be tripped. This requires two channels of the Function in the same trip system to each be OPERABLE or in trip, and the recirculation pump MG set drive motor field breakers to be OPERABLE or in trip. 3

The 72 hour Completion Time is sufficient for the operator to take corrective action (e.g., restoration or tripping of channels) and takes into account the likelihood of an event requiring actuation of the ATWS-RPT instrumentation during this period and that one Function is still maintaining ATWS-RPT trip capability.

C.1

Required Action C.1 is intended to ensure that appropriate Actions are taken if multiple, inoperable, untripped channels within both Functions result in both Functions not maintaining ATWS-RPT trip capability. The description of a Function maintaining ATWS-RPT trip capability is discussed in the Bases for Required Action B.1 above.

The 1 hour Completion Time is sufficient for the operator to take corrective action and takes into account the likelihood of an event requiring actuation of the ATWS-RPT instrumentation during this period.

BASES

ACTIONS (continued)

D.1 and D.2

With any Required Action and associated Completion Time not met, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours (Required Action D.2). Alternately, the associated recirculation pump may be removed from service since this performs the intended function of the instrumentation (Required Action D.1). The allowed Completion Time of 6 hours is reasonable, based on operating experience, both to reach MODE 2 from full power conditions and to remove a recirculation pump from service in an orderly manner and without challenging plant systems.

Required Action D.1 is modified by a Note which states that the Required Action is only applicable if the inoperable channel is the result of an inoperable [RPT] breaker. The Note clarifies the situations under which the associated Required Action would be the appropriate Required Action. (1)

SURVEILLANCE REQUIREMENTS

~~REVIEWER'S NOTE~~
~~Certain Frequencies are based on approved topical reports. In order for a licensee to use these times, the licensee must justify the Frequencies as required by the staff Safety Evaluation Report for the topical report.~~ (4)

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into the associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 3 [2]) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the recirculation pumps will trip when necessary. (3)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.2.1 1 1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the required channels of this LCO. INSERT 5 1

SR 3.3.4.2.2 1 1

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the ~~entire~~ channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. 2

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 2. 3
3

1

INSERT 5

A Note has been added to SR 3.3.4.1.1 that states the CHANNEL CHECK is not required for the time delay portion of the Reactor Vessel Water Level - Low Low Function.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.2.3 1 1

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in SR 3.3.4.2.4. If 1.5 1 the trip setting is discovered to be less conservative than the setting accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 2. 3 3

SR 3.3.4.1.4 and SR 3.3.4.2.4 1 5 1

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

INSERT 6

The Frequency of SR 3.3.4.1.5 is based upon the assumption of an 18 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. 1

SR 3.3.4.2.5 1 6 1

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the pump breakers is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a breaker is incapable of operating, the associated instrument channel(s) would be inoperable.

1 **INSERT 6**

The Frequency of SR 3.3.4.1.4 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

BASES

SURVEILLANCE REQUIREMENTS (continued)

24

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. 1

24

REFERENCES

U

1. FSAR, ~~Figure [ ] ATWS-RPT Logic Diagram~~. Section, 7.6.2.2 3 5

2. USAR, Section 14.8. 3

GENE -A

3 2. GEDE-770-06-1, "Bases for Changes To Surveillance Test Intervals and Allowed Out-of-Service Times For Selected Instrumentation Technical Specifications," ~~February 1991~~. December 1992 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.4.1 BASES, ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP (ATWS-RPT) INSTRUMENTATION

1. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific methodology, nomenclature, number, reference, system description, analysis description, or licensing basis description.

4. This Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained on the final version of the plant-specific submittal.

5. The brackets have been removed and the proper plant specific information/value has been provided.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.4.1, ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP (ATWS-RPT) INSTRUMENTATION

There are no specific NSHC discussions for this Specification.

ATTACHMENT 8

ITS 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation

Current Technical Specification (CTS) Markup and Discussion of Changes (DOCs)

A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3.2 PROTECTIVE INSTRUMENTATION	**4.2 PROTECTIVE INSTRUMENTATION**
Applicability: Applies to the plant instrumentation which performs a protective function.	Applicability: Applies to the surveillance requirements of the instrumentation that performs a protective function.
Objective: To assure the operability of protective instrumentation.	Objective: To specify the type and frequency of surveillance to be applied to protective instrumentation.
Specification:	Specification:
A. Primary Containment Isolation Functions When primary containment integrity is required, the limiting conditions of operation for the instrumentation that initiates primary containment isolation are given in Table 3.2.1.	The instrumentation to be functionally tested and calibrated and the frequency of the tests is given in Table 4.2.1.

See ITS 3.3.6.1

Note 1 to Surveillance Requirements

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A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
B. Emergency Core Cooling Subsystems Actuation When irradiated fuel is in the reactor vessel and the reactor water temperature is above 212°F, the limiting conditions for operation for the instrumentation which initiates the emergency core cooling subsystems are given in Table 3.2.2.	M.1 L.1
C. Control Rod Block Actuation 1. SRM, IRM, APRM and Scram Discharge Volume Rod Blocks The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.3. 2. Rod Block Monitor (RBM) a. When core thermal power is greater than or equal to 30% of rated and MCPR is below the limits specified in the Core Operating Limits Report, either: (1) Both RBM channels shall be operable, or (2) With one RBM channel inoperable, control rod withdrawal shall be blocked within 24 hours, or (3) With both RBM channels inoperable, control rod withdrawal shall be blocked immediately.	C. Control Rod Block Actuation During operation requiring RBM operability when only one channel is operable, an instrument functional test of the operable RBM shall be performed within 24 hours prior to withdrawal of control rod(s).

Applicability

LCO 3.3.5.1

See ITS 3.3.2.1

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A.1

ITS

A.8

A.9

LA.1

Table 3.3.5.1-1

A.2

REQUIRED CHANNELS PER FUNCTION

Table 3.2.2
Instrumentation That Initiates Emergency Core Cooling Systems

ITS	Function	Trip Setting (Allowable Value)	Minimum No. of Operable or Operating Trip Systems (3) (6)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (3) (6)	Required Conditions*
	A. Core Spray and LPCI					
	1. Pump Start					
1.a, 2.a	a. Low Low Reactor Water Level and	≥ -48" 397	2	4(4)	4	A.
1.d, 2.d	b. i. Reactor Low Pressure Permissive or	≥ 450 psig 2	2	2(4)	2	A.
1.e, 2.e	ii. Reactor Low Pressure Permissive Bypass Timer	20 ± 1 min 397 psig and ≤ 440 psig	2	1	1 2	B.
1.b, 2.b	c. High Drywell Pressure (7)	≤ 2 psig	2	4(4)	4	A.
1.c, 2.c	2. Low Reactor Pressure (Valve Permissive)	≥ 450 psig	2	2(4)	2	A.
	3. Loss of Auxiliary Power	------	2	2(2)	2	A.

L.5

M.6

A.3

A.2

A.3

LA.2

Add proposed ITS 3.3.5.1 Functions 1.f, 2.f, 2.g, 2.h, 2.i, 2.j, 2.k, 2.l, and 2.m

M.2

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A.1

A.8

LA.1

A.2

Allowable Value

REQUIRED CHANNELS PER FUNCTION

Table 3.3.5.1-1

Table 3.2.2
Instrumentation That Initiates Emergency Core Cooling Systems

	Function	~~Trip Setting~~ Allowable Value	~~Minimum No. of Operable or Operating Trip Systems (3) (6)~~	~~Total No. of Instrument Channels Per Trip System~~	~~Minimum No. of Operable or Operating Instrument Channels Per Trip System (3) (6)~~ REQUIRED CHANNELS PER FUNCTION	Required Conditions*
	B. HPCI System					
3.b	1. High Drywell Pressure ~~(1)~~ (M.6)	≤ 2 psig	1	4	4	A.
3.a	2. Low-Low Reactor Water Level	≥ -48"	1	4	4	A.
	C. Automatic Depressurization (A.2)					
4.a, 5.a	1. Low-Low Reactor Water Level and	≥ -48"	2	2	2	B.
4.b, 5.b	2. Auto Blowdown Timer and	≤ 120 seconds	2	1	1	B.
4.c, 4.d, 5.c, 5.d	3. Low Pressure Core Cooling Pumps Discharge Pressure Interlock	~~≥ 60 psig ≤ 150 psig~~ ≥ 75 psig and ≤ 125 psig	2	12(4)	~~12(4)~~ 2 for Core Spray 4 for LPCI	B.

L.2

Add proposed ITS 3.3.5.1 Functions 3.e and 3.f

M.2

M.8

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ITS

A.1

A.8

LA.1

{See ITS 3.3.8.1}

Allowable Value

REQUIRED CHANNELS PER FUNCTION

A.2

Table 3.3.5.1-1

Table 3.2.2 - (Continued)
Instrumentation That Initiates Emergency Core Cooling Systems

ITS	Function	~~Trip Setting~~ Allowable Value	~~Minimum No. of Operable or Operating Trip Systems (3) (6)~~	~~Total No. of Instrument Channels Per Trip System~~	~~Min. No. of Operable or Operating Instrument Channels Per Trip System (3) (6)~~ REQUIRED CHANNELS PER FUNCTION	Required Conditions*
Footnote (a)	D. Diesel Generator					
	1. Degraded or Loss of Voltage Essential Bus (5)					
1.a	2. Low Low Reactor Water Level	≥-48"	~~2~~	~~4(4)~~	4	C.
1.b	3. High Drywell Press	≤2 psig	~~2~~	~~4(4)~~	4	C.

A.3

NOTES:

1. ~~High drywell pressure may be bypassed when necessary only by closing the manual containment isolation valves during purging for containment inerting or de-inerting. Verification of the bypass condition shall be noted in the control room log. Also need not be operable when primary containment integrity is not required.~~ M.6

2. ~~One instrument channel is a circuit breaker contact and the other is an undervoltage relay.~~ LA.2

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A.1

ITS

Table 3.2.2 - Continued

Notes:

Add proposed ACTIONS Note — L.3

ACTION A — 3. Upon discovery that minimum requirements for the number of operable or operating trip systems, or instrument channels are not satisfied action shall be initiated as follows:

ACTIONS B, C, F, G — (a) With one required instrument channel inoperable per trip function, place the inoperable channel or trip system in the tripped condition within 12 hours or — M.7, L.3

(b) With more than one instrument channel per trip system inoperable, ~~immediately~~ 1 hour satisfy the requirements by placing appropriate channels or systems in the tripped condition, or — L.3

ACTION H — (c) Place the plant under the specified required conditions using normal operating procedures.

4. ~~All instrument channels are shared by both trip systems.~~ — LA.1

5. See table 3.2.6. — See ITS 3.3.8.1

Surveillance Requirements Note 2 — 6. A channel ~~(a shared channel is considered one channel)~~ may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided that at least one other operable channel in the same trip system is monitoring that parameter. — LA.1

* Required conditions when minimum conditions for operation are not satisfied.

ACTION H — A. Comply with Specification 3.5.A.

ACTION H — B. ~~Reactor pressure ≤ 150 psig.~~ — Declare associated support feature(s) inoperable — L.4

ACTION H — C. Comply with Specification 3.9.B.

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A.1

ITS

Table 3.3.5.1-1

SR 3.3.5.1.2, SR 3.3.5.1.7

SR 3.3.5.1.3, SR 3.3.5.1.4, SR 3.3.5.1.7

SR 3.3.5.1.1

Table 4.2.1
Minimum Test and Calibration Frequency for Core Cooling, Rod Block and Isolation Instrumentation

ITS	Instrument Channel	Test ~~(3)~~	Calibration ~~(3)~~	Sensor Check ~~(3)~~
	ECCS INSTRUMENTATION			
1.a, 2.a, 3.a, 4.a, 5.a	1. Reactor Low-Low Water Level	Once/3 months ~~(Note 5)~~ -2	~~Every~~ Once/ Operating Cycle 24 months - Transmitter -7 Once/3 months -Trip Unit -3	Once/12 hours -1
1.b, 2.b, 3.b	2. Drywell High Pressure	Once/3 months -2	Once/3 months -4	None
1.d, 2.d	3. Reactor Low Pressure (Pump Start)	Once/3 months -2	Once/3 months -4	None
1.c, 2.c	4. Reactor Low Pressure (Valve Permissive)	Once/3 months -2	Once/3 months -4	None
	~~5. Undervoltage Emergency Bus~~	~~Refueling Outage~~ Interval	~~Refueling Outage~~ Interval	~~None~~
4.c, 4.d, 5.c, 5.d	6. Low Pressure Core Cooling Pumps Discharge Pressure Interlock	Once/3 months -2	Once/3 months -4	None
	~~7. Loss of Auxiliary Power~~	~~Refueling Outage~~ Interval	~~Refueling Outage~~ Interval	~~None~~
3.d	8. Condensate Storage Tank Level	Refueling Outage Interval -7	Refueling Outage Interval -7	None
3.c	9. Reactor High Water Level	Once/3 months ~~(Note 5)~~ -2	~~Every~~ Once/ Operating Cycle 24 months - Transmitter -7 ~~Every~~ Once/ 3 months - Trip Unit -3	Once/12 hours -1
	ROD BLOCKS			
	1. APRM Downscale	Once/3 months (Note 5)	Once/3 months	None
	2. APRM Flow Variable	Once/3 months (Note 5)	Once/3 months	None
	3. IRM Upscale	Notes (2,5)	Note 2	Note 2
	4. IRM Downscale	Notes (2,5)	Note 2	Note 2
	5. RBM Upscale	Once/3 months (Note 5)	Once/3 months	None
	6. RBM Downscale	Once/3 months (Note 5)	Once/3 months	None
	7. SRM Upscale	Notes (2,5)	Note 2	Note 2
	8. SRM Detector Not-Full-In Position	Notes (2,9)	Note 2	None
	9. Scram Discharge Volume-High Level	Once/3 months	Refueling Outage	None
	MAIN STEAM LINE (GROUP 1) ISOLATION			
	1. Steam Tunnel High Temperature	Refueling Outage	Refueling Outage	None
	2. Steam Line High Flow	Once/3 months	Once/3 Months	Once/12 hours

A.9

INSERT 1

A.7

A.9

A.9

A.9

A.7

A.9

A.6

A.5

Add proposed Notes (c) and (d) to Table 3.3.5.1-1

L.5

LA.2

M.8

Add proposed Notes (c) and (d) to Table 3.3.5.1-1

LA.2

A.5

See ITS 3.3.2.1

See ITS 3.3.6.1

Add proposed SR 3.3.5.1.8

M.4

Amendment No. ~~2~~, ~~10~~, ~~37~~, ~~39~~, ~~63~~, ~~66~~, ~~81~~, ~~103~~, 104

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A.1

ITS

A.9 INSERT 1

Table 3.3.5.1-1

1.e, 2.e	10.	Reactor Low Pressure (Bypass Timer)	Refueling Interval -7	Refueling Interval -7	24 months	None
4.b, 5.b	11.	Auto Blowdown Timer	Refueling Interval -7	Refueling Interval -7		None

A.4

A.5

Insert Page 61

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A.1

Table 4.2.1 Continued
Minimum Test and Calibration Frequency for Core Cooling,
Rod Block and Isolation Instrumentation

NOTES:

(1) (Deleted)

(2) Calibrate prior to normal shutdown and start-up and thereafter check once per 12 hours and test once per week until no longer required. Calibration of this instrument prior to normal shutdown means adjustment of channel trips so that they correspond, within acceptable range and accuracy, to a simulated signal injected into the instrument (not primary sensor). In addition, IRM gain adjustment will be performed, as necessary, in the APRM/IRM overlap region. See ITS 3.3.2.1

(3) Functional tests, calibrations ~~and~~ sensor checks are not required when the systems are ~~not~~ required to be operable or are tripped. If tests are missed, ~~they shall~~ be performed prior to returning the systems to ~~an operable~~ status. A.6

(4) Whenever fuel handling is in process, a sensor check shall be performed once per 12 hours. See ITS 3.3.6.2

(5) A functional test of this ~~instrument means~~ the injection of a simulated signal into the ~~instrument~~ (not primary sensor) to verify the proper instrument ~~channel~~ response alarm and/or initiating action. A.7

(6) (Deleted)

(7) (Deleted)

(8) Once/shutdown if not tested during previous 3 month period. See ITS 3.3.6.3

(9) Testing of the SRM Not-Full-In rod block is not required if the SRM detectors are secured in the full-in position. See ITS 3.3.2.1

(10) Uses contacts from scram system. Tested and calibrated in accordance with Tables 4.1.1 and 4.1.2. See ITS 3.3.6.1 and ITS 3.3.6.2

(11) Uses contacts from Group 1 isolation logic. Tested and calibrated in accordance with Group 1 Low Low Water Level Instrumentation.

(12) Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. See ITS 3.3.6.1

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A.1

ITS

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. Rod Block Monitor (RBM) (continued) b. RBM Setpoints for control rod block are given in Table 3.2.3. The upscale LTSP shall be applied above 30% and up to 65% of rated thermal power. The upscale ITSP shall be applied at and above 65% and up to 85% of rated thermal power. The upscale HTSP shall be applied at and above 85% of rated thermal power. The RBM Bypass time delay shall be less than or equal to 2.0 seconds.	See ITS 3.3.2.1
D. Other Instrumentation	M.5
Applicability, Table 3.3.5.1-1 Functions 3.c and 3.d — Whenever the reactor is in the RUN Mode, the limiting conditions for operation for the instrumentation listed in Table 3.2.8 shall be met. (LCO 3.3.5.1)	

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A.1

ITS

A.8

LA.1

Allowable Value

A.2

REQUIRED CHANNELS PER FUNCTION

Table 3.2.8 Other Instrumentation					
Function	Trip Setting	Minimum No. of Operable or Operating Trip System (1) (2)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (1) (2)	Required Conditions*
A. RCIC Initiation					
1. Low-Low Reactor Level	≥-48"	1	4	4	B
B. HPCI/RCIC Turbine Shutdown					
3.c 1. High Reactor Level	≤48"	1	2	2	A
C. HPCI/RCIC Turbine Suction Transfer					
3.d 1. Condensate Storage Tank Low Level Allowable Values	≥2/3" above tank bottom (Two Tank Operation)	1	2	2	C
	≥6/9" above tank bottom (One Tank Operation)	1	2	2	C

See ITS 3.3.5.2

29.3 inches

LA.3

L.5

LA.3

See ITS 3.3.5.2

LA.1

Add proposed ACTIONS Note

L.3

NOTE:

ACTION A 1. Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied, action shall be initiated as follows:

ACTIONS C and D

a. With one required instrument channel inoperable per trip function, place the inoperable channel or trip system in the tripped condition within 12 hours, or 24

M.7

b. With more than one instrument channel per trip system inoperable, immediately satisfy the requirements by placing the appropriate channels or systems in the tripped condition, or

L.3

1 hour

ACTION H c. Place the plant under the specified required condition using normal operating procedures.

Surveillance Requirements Note 2 2. A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided that at least one other operable channel in the same trip system is monitoring that parameter.

* Required conditions when minimum conditions for operation are not satisfied:

ACTION H A. Comply with Specification 3.5.A.

B. Comply with Specification 3.5.D.

See ITS 3.3.5.2

Required Action D.2.2 C. Align HPCI and RCIC suction to the suppression pool.

Add proposed ACTION H

M.3

3.2/4.2

See ITS 3.3.5.2

60d 06/11/02

Amendment No. ~~37, 93, 103, 105,~~ 128

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DISCUSSION OF CHANGES
ITS 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

ADMINISTRATIVE CHANGES

A.1 In the conversion of the Monticello Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS Table 3.2.2 and CTS Table 3.2.8 specify the "Minimum No. of Operable or Operating Instrument Channels Per Trip System." ITS Table 3.3.5.1-1 specifies the "REQUIRED CHANNELS PER FUNCTION." This changes the CTS by changing the title of the "Minimum No. of Operable or Operating Instrument Channels Per Trip System" column to "REQUIRED CHANNELS PER FUNCTION." Due to this change, the number of channels required has been increased by multiplying the number of trip systems (specified in the "Minimum No. of Operable or Operating Trip Systems" column) by the number of required channels per trip system (specified in the "Minimum No. of Operable or Operating Instrument Channels Per Trip System" column). For Functions where this is not the case, a specific DOC justifying the change has been provided. However, the ADS Instrumentation has been split into two Functions; Trip System A and Trip System B instrumentation. Therefore, the number of channels required has not been changed.

The purpose of CTS Table 3.2.2, in part, is to identify the number of channels per trip system and the number of trip systems. All channels provided by the design are currently required to be OPERABLE. This change is acceptable because the "REQUIRED CHANNELS PER FUNCTION" reflects, for the most part, the total number of channels available for each Function. Any changes to these requirements are identified below. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS Table 3.2.2 Functions A.1.a, A.1.c, D.2, and D.3 (Low Low Reactor Water Level and High Drywell Pressure) require four channels per trip system to be OPERABLE. CTS Table 3.2.2 Functions A.1.b.i and A.2 (Reactor Low Pressure Permissive and Low Reactor Pressure (Valve Permissive)) require two channels per trip system to be OPERABLE. For all four of these Functions, CTS Table 3.2.2 states that there are two trip systems. Furthermore, CTS Table 3.2.2 Note 4, which applies to each of these Functions, states that all instrument channels are shared by both trip systems, and CTS Table 3.2.2 Note 6, which applies to each of these Functions, states, in part, that a shared channel is considered one channel. ITS Table 3.3.5.1-1 Functions 1.a and 2.a (Reactor Vessel Water Level - Low Low) and Functions 1.b and 2.b (Drywell Pressure - High) require four channels per Function to be OPERABLE and ITS Table 3.3.5.1-1 Functions 1.c and 2.c (Reactor Steam Dome Pressure - Low (Injection Permissive) and Functions 1.d and 2.d (Reactor Steam Dome Pressure - Low (Pump Permissive)) require two channels per Function to be OPERABLE. This changes the CTS by clarifying the actual number of channels required to be OPERABLE on a per Function basis.

DISCUSSION OF CHANGES
ITS 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

The purpose of specifying the number of channels required to be OPERABLE for each Function is to ensure the ECCS instrumentation can perform its assumed function. The change to the number of channels is only the result of converting to the NUREG-1433, Revision 3 terminology. The CTS Table Notes clearly state that while either two or four channels are required by each trip system, the channels for one trip system are common and shared with the other trip system. Therefore, since each channel inputs into both trip systems, the total number of channels required per Function is not twice as many as is required on a per trip system basis, but is the same number. Therefore, this change is acceptable and is simply a presentation preference. This change is designated as administrative because it does not result in technical changes to the CTS.

A.4 CTS Table 4.2.1 ECCS Instrument Channel 8, Instrument Channel 10, and Instrument Channel 11 require the performance of a CHANNEL FUNCTIONAL TEST and a CHANNEL CALIBRATION at the same Frequency (Refueling Interval). ITS Table 3.3.5.1-1 Functions 1.e, 2.e, 3.d, 4.b, and 5.b only require the performance of a CHANNEL CALIBRATION at a similar Frequency. This changes the CTS by combining the current CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION testing requirements into a single Surveillance Requirement.

ITS 1.1, "Definitions," includes the definition of CHANNEL CALIBRATION. It states that the CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY "and the CHANNEL FUNCTIONAL TEST." Therefore, the CHANNEL CALIBRATION test in the ITS will encompass all aspects of the CHANNEL FUNCTIONAL TEST. This change is acceptable because there are no changes in the testing requirements. Therefore, requiring a CHANNEL FUNCTIONAL TEST Surveillance at the same Frequency as a CHANNEL CALIBRATION is redundant and unnecessary. This change is designated as administrative because it does not result in technical changes to the CTS.

A.5 CTS Table 4.2.1 ECCS Instrument Channels 1 and 9 specify an "Operating Cycle" Frequency for the CHANNEL CALIBRATION requirement while CTS Table 4.2.1 ECCS Instrument Channels 8, 10, and 11 specify a "Refueling Interval" Frequency for both the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION requirements. ITS SR 3.3.5.1.7 requires the performance of a CHANNEL CALIBRATION every "24 months." This changes the CTS by changing the Frequency from once per "Operating Cycle" or "Refueling Interval" to "24 months."

This change is acceptable because the current "Operating Cycle" and "Refueling Interval" are "24 months." In letter L-MT-04-036, from Thomas J. Palmisano (NMC) to the USNRC, dated June 30, 2004, NMC has proposed to extend the fuel cycle from 18 months to 24 months and at the same time has performed an evaluation in accordance with Generic Letter 91-04 to extend the unit Surveillance Requirements from 18 months to 24 months. CTS Table 4.2.1 was included in this evaluation. This change is designated as administrative because it does not result in any technical changes to the CTS.

DISCUSSION OF CHANGES
ITS 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

A.6 CTS Table 4.2.1 Note (3) states that functional tests, calibrations, and sensor checks are not required when the systems are not required to be OPERABLE or are tripped. In addition, the Note states that if tests are missed, they shall be performed prior to returning the systems to an OPERABLE status. These explicit requirements are not retained in ITS 3.3.5.1. This changes the CTS by not including these explicit requirements.

The purpose of this Note is to provide guidance on when Surveillances are required to be met and performed. This explicit Note is not needed in ITS 3.3.5.1 since these allowances are included in ITS SR 3.0.1. ITS SR 3.0.1 states that SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR, and failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. SR 3.0.1 also states that SRs are not required to be performed on inoperable equipment. When equipment is declared inoperable, the Actions of this LCO require the equipment to be placed in the trip condition. In this condition, the equipment is still inoperable but has accomplished the required safety function. Therefore the allowances in SR 3.0.1 and the associated actions provide adequate guidance with respect to when the associated surveillances are required to be performed and this explicit requirement is not retained. This change is administrative because it does not result in a technical change to the CTS.

A.7 CTS Table 4.2.1 Note (5) states that a functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response, alarm, and/or initiating action. These explicit requirements are not retained in ITS 3.3.5.1. This changes the CTS by not including these explicit requirements.

The purpose of CTS Table 4.2.1 Note (5) is to provide guidance on how to perform an instrument functional test of the ECCS Reactor Low Low Water Level and Reactor High Water Level instrument channels. This explicit Note is not needed in ITS 3.3.5.1 since the requirements for the CHANNEL FUNCTIONAL TEST are included in ITS 1.1, "Definitions." ITS 1.1 states that a CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. Therefore, the ITS 1.1 definition provides adequate guidance with respect to performance requirements of a CHANNEL FUNCTIONAL TEST and this explicit requirement is not retained. This change is administrative because it does not result in a technical change to the CTS.

A.8 CTS 3.2.B states that the limiting conditions for operation for the instrumentation which initiates ECCS are given in Table 3.2.2. CTS Table 3.2.2 specifies the "Trip Setting" for each ECCS Function. CTS 3.2.D states that the limiting conditions for operation for the instrumentation listed in Table 3.2.8 shall be met. CTS Table 3.2.8 specifies the "Trip Setting" for HPCI actuation Functions. ITS LCO 3.3.5.1 requires the ECCS instrumentation for each Function in Table 3.3.5.1-1 to be OPERABLE and ITS Table 3.3.5.1-1 specifies the "Allowable Value" for each Function. This changes the CTS by replacing the term "Trip Setting" with "Allowable Value."

DISCUSSION OF CHANGES
ITS 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

The purpose of the "Trip Setting" in CTS Table 3.2.2 and CTS Table 3.2.8 is to define the OPERABILITY limits for the ECCS instrumentation Functions. Therefore, the use of the term "Trip Setting" in the CTS is the same as the use of the term "Allowable Value" in the ITS. This proposed change does not modify the actual "Trip Setting" specified in CTS Tables 3.2.2 or 3.2.8 for the ECCS Functions. Any changes to the actual value specified in the "Trip Setting" (i.e., changing the value for OPERABILITY) are discussed in DOCs M.8 and L.5. This change is designated as administrative change and is acceptable because it does not result in any technical changes to the CTS.

A.9 These changes to CTS Table 3.2.2 and CTS Table 4.2.1 are provided in the Monticello ITS consistent with the Technical Specifications Change Request submitted to the USNRC for approval in NMC letter L-MT-04-036, from Thomas J. Palmisano (NMC) to USNRC, dated June 30, 2004. As such, these changes are administrative.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.2.B requires the ECCS Instrumentation Functions in Table 3.2.2 to be OPERABLE when irradiated fuel is in the reactor vessel and the reactor water temperature is above 212°F. ITS Table 3.3.5.1-1 requires the low pressure ECCS Instrumentation (ITS Table 3.3.5.1-1 requirements for Core Spray (CS) and Low Pressure Coolant Injection (LPCI) System) Functions to be OPERABLE in MODES 1, 2, and 3. In addition some Functions (ITS Table 3.3.5.1-1 Functions 1.a, 1.c, 1.d, 1.e, 1.f, 2.a, 2.c, 2.d, and 2.e) are required to be OPERABLE in MODE 4 and 5 when the associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2, "ECCS - Shutdown." This changes the CTS by requiring the ECCS Instrumentation Functions to be OPERABLE in MODE 2 when reactor water temperature is less than or equal to 212°F and in MODE 4 and 5 when the associated ECCS subsystem(s) are required to be OPERABLE.

The purpose of CTS 3.2.B and CTS Table 3.2.2 are to ensure the ECCS Instrumentation is OPERABLE to mitigate the consequences of a design basis accident. ECCS Instrumentation is required to be OPERABLE during MODES 1, 2, and 3 when a design basis accident could cause a release of radioactive material to the primary containment. In MODES 1 and 3, the reactor coolant temperature will always be above 212°F. In MODE 2, the reactor coolant temperature may be less than or equal to 212°F when the reactor is subcritical but control rods are withdrawn. Therefore, it is necessary and acceptable to require the ECCS Instrumentation to be OPERABLE. In addition, the low pressure ECCS are required to be OPERABLE in MODES 4 and 5 whenever the associated ECCS subsystem(s) are required to be OPERABLE. This will ensure the associated low pressure ECCS subsystems can start automatically when required in order to mitigate the consequences of a vessel draindown. This change is designated as more restrictive because the LCO will be applicable under more reactor operating conditions than in the CTS.

DISCUSSION OF CHANGES
ITS 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

M.2 CTS Table 3.2.2 and Table 4.2.1 include requirements for ECCS Instrumentation. The Table does not include the requirements for the Core Spray Pump Start - Time Delay, LPCI Pump Start - Time Delay, LPCI Pump Discharge Flow - Low (Bypass), LPCI Loop Select Logic, HPCI Suppression Pool Water Level - High, and HPCI Pump Discharge Flow - Low (Bypass) Functions. ITS LCO 3.3.5.1 and Table 3.3.5.1-1 Functions 1.f, 2.f, 2.g, 2.h, 2.i, 2.j, 2.k, 2.l, 2.m, 3.e, and 3.f have been added to ensure the applicable ECCS Instrumentation is OPERABLE. Appropriate ACTIONS and Surveillances have been also added. This changes the CTS by adding the requirements for the specified Functions.

The purpose of these Functions is to ensure the ECCS Instrumentation necessary to support the safety analysis is OPERABLE. This change is acceptable because the added ECCS Instrumentation Functions will help to support the safety analyses. This change is designated as more restrictive because it adds an explicit LCO, Applicability, ACTIONS, and Surveillance Requirements for the specified ECCS Instrumentation Functions to the Technical Specifications.

M.3 CTS Table 3.2.8 Function C.1 (Condensate Storage Tank Low Level) requires entry into Required Condition C when the requirements of CTS Table 3.2.8 Notes 1.a and 1.b are not met. CTS Table 3.2.8 Required Condition C requires the HPCI suction to be aligned to the suppression pool. CTS Table 3.2.8 does not provide an alternate compensatory action if the HPCI suction cannot be aligned to the suppression pool. Under this condition, ITS 3.3.5.1 ACTION H requires the immediate declaration that the HPCI System is inoperable. This changes the CTS by providing an appropriate compensatory action if the Required Action is not met.

The purpose of CTS Table 3.2.8 Required Condition C is to align the HPCI pump suction to the suppression pool when the Condensate Water Level - Low requirements are not met, since this performs the function of the instrumentation. CTS Table 3.2.8 does not provide an alternate compensatory action when this action is not met. ITS 3.3.5.1 ACTION H requires the immediate declaration that the HPCI System is inoperable. The Condensate Water Level - Low Function supports the OPERABILITY of the HPCI System. If any requirement associated with HPCI System Instrumentation is not met, declaring the HPCI System inoperable is the appropriate action because the instrumentation supports the automatic operation of the HPCI System. The change is acceptable and necessary because it provides the appropriate compensatory action when the HPCI suction cannot be aligned to the suppression pool. This change is designated as more restrictive because it adds a specific requirement to declare the HPCI System inoperable when the HPCI System suction cannot be aligned to the suppression pool.

M.4 CTS Table 4.2.1 does not specify requirements for a LOGIC SYSTEM FUNCTIONAL TEST. ITS Table 3.3.5.1-1 requires the performance of SR 3.3.5.1.8, a LOGIC SYSTEM FUNCTIONAL TEST, every 24 months, for each ECCS Function. This changes the CTS by explicitly requiring a LOGIC SYSTEM FUNCTIONAL TEST to be performed on each ECCS Function.

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This change is acceptable because a LOGIC SYSTEM FUNCTIONAL TEST helps to ensure the ECCS logic is functioning as required to support the safety analyses. As such, explicitly including requirements for a LOGIC SYSTEM FUNCTIONAL TEST in the Technical Specifications provides additional assurance that the OPERABILITY of the ECCS Instrumentation Functions will be maintained. This change is designated as more restrictive because it adds a specific requirement to perform a LOGIC SYSTEM FUNCTIONAL TEST on each ECCS Instrumentation Function.

M.5 CTS 3.2.D requires the specified HPCI Functions in Table 3.2.8 (HPCI High Reactor Level (Function B.1) and HPCI Condensate Storage Tank Low Level (Function C.1)) to be OPERABLE in the Run Mode. ITS Table 3.3.5.1-1 Functions 3.c and 3.d require the Functions to be OPERABLE in MODE 1 and MODES 2 and 3 with reactor steam dome pressure > 150 psig. This changes the CTS by requiring the HPCI Instrumentation Functions to be OPERABLE in MODES 2 and 3 with reactor steam dome pressure > 150 psig.

The purpose of CTS 3.2.D and CTS Table 3.2.8 is to ensure the HPCI Instrumentation is OPERABLE to mitigate the consequences of a design basis accident. HPCI Instrumentation is required to be OPERABLE to support the operation of the HPCI System. The HPCI System is currently required to be OPERABLE when reactor steam dome pressure is > 150 psig (CTS 3.5.A.2). This requirement is retained in ITS 3.5.1. Therefore, it is appropriate that the HPCI Instrumentation is OPERABLE when the HPCI System is required to be OPERABLE. This change is acceptable because it aligns the Applicability of the HPCI instruments with the supported equipment. This change is designated as more restrictive because the LCO will be applicable under more reactor operating conditions than in the CTS.

M.6 CTS Table 3.2.2 Note 1 allows the ECCS High Drywell Pressure Functions (Functions A.1.c and B.1) to be bypassed during purging for containment inerting or de-inerting operations by closing the manual containment isolation valves. ITS Table 3.3.5.1-1 does not include this bypass allowance for the Drywell Pressure - High Functions (Functions 1.b, 2.b, and 3.b). This changes the CTS by deleting the allowance to bypass the Drywell Pressure - High Functions during containment purging operations.

The purpose of this Note is to allow the High Drywell Pressure Functions to be bypassed to avoid an inadvertent actuation of the ECCS during the containment purging operations. Monticello does not utilize this High Drywell Pressure Function bypass allowance during any type of purging operation. The allowance has been deleted from the Technical Specifications. This change is acceptable because this allowance is not needed for purging operations. This change is designated as more restrictive because the bypass allowance has been deleted from the Technical Specifications.

M.7 CTS Table 3.2.2 Note 3.(a) applies, in part, to the Reactor Low Pressure Permissive (Function A.1.b.i), Reactor Low Pressure Permissive Bypass Timer (Function A.1.b.ii), Low Reactor Pressure (Valve Permissive) (Function A.2), Auto Blowdown Timer (Function C.2), and Low Pressure Core Cooling Pumps Discharge Pressure Interlock (Function C.3) channels and, when a channel is

DISCUSSION OF CHANGES
ITS 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

inoperable, requires the channel or trip system to be placed in the tripped condition within 12 hours. CTS Table 3.2.8 Note 1.a applies to the High Reactor Level channels and, when a channel is inoperable, requires the channel or trip system be placed in the tripped condition within 12 hours. ITS 3.3.5.1 ACTION C applies to ITS Table 3.3.5.1-1 Functions 1.c, 1.d, 1.e, 2.c, 2.d, 2.e, and 3.c and requires the channel to be restored to OPERABLE status within 24 hours. ITS 3.3.5.1 ACTION G applies to ITS Table 3.3.5.1-1 Functions 4.b, 4.c, 4.d, 5.b, 5.c, and 5.d and requires the channels to be restored to OPERABLE status in either 96 hours or 8 days. This changes the CTS by requiring the channel to be restored to OPERABLE status instead of requiring the inoperable channel or trip system to be placed in the tripped condition. The change in the Completion Time from 12 hours to 24 hours, 96 hours, or 8 days is discussed in DOC L.3.

The purpose of CTS Table 3.2.2 Note 3.(a) and CTS Table 3.2.8 Note 1.a is to allow a period of time before placing the channels in the tripped condition. This change does not allow the inoperable channels to be placed in the tripped condition. This change is acceptable because placing the channels in trip is not the appropriate action. For ITS Table 3.3.5.1 Functions 1.c, 1.d, 1.e, 2.c, 2.d, and 2.e, placing the channels in the tripped condition is not a desired required action because if an accident were to occur and ECCS is required to actuate, it may cause low pressure ECCS injection valves to open or pumps to start before reactor vessel pressure has decreased to below the design pressure of the low pressure ECCS piping. For ITS Table 3.3.5.1-1 Functions 4.b, 4.c, 4.d, 5.b, 5.c, and 5.d placing the channels in trip will indicate to the ADS logic that a low pressure ECCS pump is available when in fact it may not be. ITS Table 3.3.5.1-1 Function 3.c (Reactor Vessel Water Level - High) includes two channels arranged in a two-out-of-two logic. The logic will cause the HPCI turbine to trip by closing the turbine stop valve. The CTS allows the unit to operate with one HPCI Reactor Vessel Water Level - High channel in the tripped condition. Under this condition, if HPCI starts on an automatic initiation signal and is operating to mitigate the consequences of an event, a single failure of the other Reactor Vessel Water Level - High channel will make HPCI unavailable. Therefore, this requirement has been changed to require restoration of the inoperable channel. This change is more restrictive because the plant will not be able to operate indefinitely with certain channels in the tripped condition.

M.8 CTS Table 3.2.2 specifies the "Trip Setting" for the ECCS instrumentation Functions. The Trip Settings of CTS Table 3.2.2 Function C.3 has been modified to reflect new Allowable Values as indicated for ITS Table 3.3.5.1-1 Functions 4.c, 4.d, 5.c, and 5.d. In addition, two Notes have been added (ITS Table 3.3.5.1-1 Notes (c) and (d)) to the CHANNEL CALIBRATION (ITS SR 3.3.5.1.4) associated with ITS Table 3.3.5.1-1 Functions 4.c, 4.d, 5.c, and 5.d. Note (c) states, "If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service." Note (d) states, "The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the nominal trip setpoint; otherwise, the channel shall be declared inoperable. The nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified in the Technical Requirements Manual (TRM)." This changes the CTS by requiring the ECCS instrumentation to be set consistent

DISCUSSION OF CHANGES
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with the new "Allowable Value" and adds the Notes to the CHANNEL CALIBRATION (ITS SR 3.3.5.1.4) associated with ITS Table 3.3.5.1-1 Functions 4.c, 4.d, 5.c, and 5.d. The change in the term "Trip Setting" to "Allowable Value" is discussed in DOC A.8.

The purpose of the Allowable Values is to ensure the instruments function as assumed in the safety analyses. ITS Table 3.3.5.1-1 reflects Allowable Values consistent with the philosophy of General Electric ISTS, NUREG-1433. These Allowable Values have established using the GE setpoint methodology guidance, as specified in the Monticello setpoint methodology. The analytic limits are derived from limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limit. The difference between the analytic limit and the Allowable Value allows for channel instrument accuracy, calibration accuracy, process measurement accuracy, and primary element accuracy. The margin between the Allowable Value and the nominal trip setpoint (NTSP) allows for instrument drift that might occur during the established surveillance period. Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift. These two verifications are statistical evaluations to provide additional assurance of the acceptability of the NTSP and may require changes to the NTSP. Use of these methods and verifications provides the assurance that if the setpoint is found conservative to the Allowable Value during surveillance testing, the instrumentation would have provided the required trip function by the time the process reached the analytic limit for the applicable events. The Notes (ITS Table 3.3.5.1-1 Notes (c) and (d)) added to the CHANNEL CALIBRATION (ITS SR 3.3.5.1.4) associated with Functions 4.c, 4.d, 5.c, and 5.d have been added at the request of the NRC, made during the review of the original ITS submittal. Therefore, based on the above discussion, the changes to the Allowable Values are acceptable. This change is designated as more restrictive because more stringent Allowable Values are being applied in the ITS than were applied in the CTS.

<u>RELOCATED SPECIFICATIONS</u>

None

<u>REMOVED DETAIL CHANGES</u>

LA.1 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.2.2 and CTS Table 3.2.8 specify the "Minimum No. of Operable or Operating Trip Systems" and the "Total No. of Instrument Channels Per Trip System" requirements for the ECCS Instrumentation. CTS Table 3.2.2 Note 4 states that certain instrument channels are shared by both trip systems and CTS Table 3.2.2 Note 6 states that a shared channel is considered one channel. ITS 3.3.5.1 does not include these details. This changes the CTS by moving the information of the "Minimum No. of Operable or Operating Trip

DISCUSSION OF CHANGES
ITS 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

Systems," "Total No. of Instrument Channels Per Trip System," and the information that the channels are shared to the ITS Bases.

The removal of these details, which are related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the minimum number of required channels for each Function. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LA.2 *(Type 6 – Removal of LCO, SR, or other TS requirement to the TRM, UFSAR, ODCM, QAPD, or IIP)* CTS Table 3.2.2 Function 3 requires the Loss of Auxiliary Power Function to be OPERABLE. CTS Table 3.2.2 Note 2 states that the two channels associated with each trip system consists of a circuit breaker contact and an undervoltage relay. CTS Table 4.2.1 ECCS Instrumentation Function 5 requires a functional test and a calibration of the "Undervoltage Emergency Bus" channels while Function 7 requires a functional test and calibration of the "Loss of Auxiliary Power" channels. These tests are required each "Refueling Interval." These requirements are not included in ITS 3.3.5.1. This changes the CTS by moving the explicit Loss of Auxiliary Power requirements from the Technical Specifications to the Technical Requirements Manual (TRM).

The removal of these details from the Technical Specifications is acceptable because this type of information is not necessary to provide adequate protection of public health and safety. The purpose of CTS Table 3.2.2 Function 3 is to ensure the circuit breaker contacts and undervoltage relays provide the appropriate input to initiate the ECCS pump start timers so that the pumps do not start simultaneously after power is restored to the essential buses. This change is acceptable because ITS SR 3.3.5.1.8 (LOGIC SYSTEM FUNCTIONAL TEST) along with the ECCS simulated automatic actuation test specified in ITS 3.5.1 and the specified testing in ITS 3.8.1 will ensure the Loss of Auxiliary Function channels remain OPERABLE. The Surveillance Frequency of all these tests is 24 months. These requirements will ensure requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. With the removal of OPERABILITY requirements of the Loss of Auxiliary Power channels from the Technical Specification, inoperabilities associated with these Functions will be determined in accordance with associated Technical Specification system OPERABILITY requirements. Also, this change is acceptable because the removed information will be adequately controlled in the TRM. The TRM is incorporated by reference into the USAR and any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because a requirement is being removed from the Technical Specifications.

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ITS 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

LA.3 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS Table 3.2.8 specifies that the Condensate Storage Tank Low Level (Function C.1) Trip Setting is referenced from "above tank bottom." ITS Table 3.3.5.1-1 Function 3.d does not include this detail. This changes the CTS by moving the information associated with the condensate storage tank level reference point to the ITS Bases.

The removal of this detail, which is related to system design, from the Technical Specifications, is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement for the Condensate Storage Tank Level - Low Allowable Value. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 2 – Relaxation of Applicability)* CTS 3.2.B requires the ECCS Instrumentation Functions in Table 3.2.2 to be OPERABLE when irradiated fuel is in the reactor vessel and the reactor water temperature is above 212°F. ITS Table 3.3.5.1-1 requires the High Pressure Coolant Injection (HPCI) System Instrumentation and the Automatic Depressurization System (ADS) Instrumentation to be OPERABLE in MODE 1 and MODES 2 and 3 with reactor steam dome pressure is > 150 psig. This changes the CTS by deleting the requirement for the HPCI and ADS Instrumentation to be OPERABLE when the reactor water temperature is > 212°F but reactor steam dome pressure is ≤ 150 psig.

The purpose of CTS 3.2.B and CTS Table 3.2.2 are to ensure the ECCS Instrumentation is OPERABLE to mitigate the consequences of a design basis accident. CTS 3.5.A.2 only requires the HPCI and ADS Systems to be OPERABLE when reactor steam dome pressure is > 150 psig. This requirement is retained in ITS 3.5.1. This change is therefore acceptable because it aligns the Applicability of the HPCI and ADS instruments with the supported equipment. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

L.2 *(Category 1 – Relaxation of LCO Requirements)* CTS Table 3.2.2 Function C.3 requires 12 channels of Low Pressure Core Cooling Pumps Discharge Pressure Interlock to be OPERABLE in each trip system. However, CTS Table 3.2.2 Note 4, which applies to each of these Functions, states that all instrument channels are shared by both trip systems. Thus, there are 12 total channels for this Function. ITS Table 3.3.5.1-1 Function 4.c requires two Core Spray Pump Discharge Pressure - High channels to be OPERABLE for ADS Trip System A. ITS Table 3.3.5.1-1 Function 4.d requires four Low Pressure Coolant Injection Pump Discharge Pressure - High channels to be OPERABLE for ADS Trip

DISCUSSION OF CHANGES
ITS 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

System A. ITS Table 3.3.5.1-1 Function 5.c requires two Core Spray Pump Discharge Pressure - High channels to be OPERABLE for ADS Trip System B. ITS Table 3.3.5.1-1 Function 5.d requires four Low Pressure Coolant Injection Pump Discharge Pressure - High channels to be OPERABLE for ADS Trip System B. This changes the CTS by only requiring the low pressure ECCS pumps powered from the same electrical division to provide input into the associated (same electrical power division) ADS trip system.

The purpose of CTS Table 3.2.2 Function C.3 is to ensure each ADS trip system receives two pump discharge signals from each low pressure ECCS pump. This change is acceptable because the LCO requirements continue to ensure that the ADS trip system are maintained consistent with the safety analyses. These changes are based on the reliability analyses of NEDC-30936-P-A, Parts 1 and 2 dated December 1988, and found to be acceptable. The result of the NRC review of this generic reliability analyses as it relates to Monticello is documented in the NRC Safety Evaluation Report (SER) dated December 23, 1998. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

L.3 *(Category 3 – Relaxation of Completion Time)* CTS Table 3.2.2 Note 3 provides the Actions for inoperable ECCS instrumentation channels. When one instrument channel is inoperable (per trip function), CTS Table 3.2.2 Note 3.(a) requires the inoperable channel or trip system be placed in the tripped condition within 12 hours. When more than one instrument channel per trip system is inoperable, CTS Table 3.2.2 Note 3.(b) requires the immediate placement of the appropriate channels in the tripped condition. CTS Table 3.2.8 Note 1 provides the Actions for inoperable required HPCI instrumentation channels. When one required instrument channel is inoperable, CTS Table 3.2.8 Note 1.a requires the inoperable channel or trip system be placed in the tripped condition within 12 hours. When more than one instrument channel is inoperable, CTS Table 3.2.8 Note 1.b requires the immediate placement of the appropriate channels or trip system in the tripped condition. ITS Table 3.3.5.1-1 Functions 1.a, 1.b, 2.a, 2.b, 3.a, and 3.b require entry into ITS 3.3.5.1 ACTION B, which requires placement of the channel in the tripped condition within 24 hours. In addition, during operations in MODES 1, 2, and 3 (for Functions 1.a, 1.b, 2.a, and 2.b only), when its redundant feature ECCS initiation capability is inoperable, the supported feature(s) must be declared inoperable within 1 hour from discovery of loss of initiation capability for feature(s) in both divisions. In addition, for Functions 3.a and 3.b, when HPCI initiation capability is lost, the HPCI System must be declared inoperable within 1 hour from discovery of loss of HPCI initiation capability. ITS Table 3.3.5.1-1 Functions 1.c, 1.d, 1.e, 2.c, 2.d, 2.e, and 3.c require entry into ITS 3.3.5.1 ACTION C, which requires restoration of the channel within 24 hours. In addition, during operations in MODES 1, 2, and 3 (for Functions 1.c, 1.d, 1.e, 2.c, 2.d, and 2.e), when its redundant feature ECCS initiation capability is inoperable, the supported feature(s) must be declared inoperable within 1 hour from discovery of loss of initiation capability for feature(s) in both divisions. ITS Table 3.3.5.1-1 Function 3.d requires entry into ITS 3.3.5.1 ACTION D, which requires either the placement of the channel in the tripped condition or the alignment of the HPCI pump suction to the suppression pool within 24 hours. In addition, when both channels are inoperable, the HPCI System must be declared inoperable within 1 hour from discovery of loss of HPCI

DISCUSSION OF CHANGES
ITS 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

initiation capability (if the HPCI pump suction is not aligned to the suppression pool). ITS Table 3.3.5.1-1 Functions 4.a and 5.a require entry into ITS 3.3.5.1 ACTION F, which requires placement of the channel in the tripped condition within 96 hours from discovery of the inoperable channel concurrent with HPCI or reactor core isolation cooling (RCIC) inoperable and within 8 days (if both HPCI and RCIC are OPERABLE). In addition the ADS valves must be declared inoperable within 1 hour from discovery of loss of ADS initiation capability in both trip systems. ITS Table 3.3.5.1-1 Functions 4.b, 4.c, 4.d, 5.b, 5.c, and 5.d require entry into ITS 3.3.5.1 ACTION G, which requires restoration of the channel to OPERABLE status within 96 hours from discovery of inoperable channel concurrent with HPCI or RCIC inoperable and within 8 days (if both HPCI and RCIC are OPERABLE). In addition the ADS valves must be declared inoperable within 1 hour from discovery of loss of ADS initiation capability in both trip systems. Finally, a Note has been added to the ACTIONS that states that separate Condition entry is allowed for each channel. This changes the CTS by extending the time allowed to take action for each individual channel, as long as ECCS initiation capability is maintained. Specifically, it extends the Completion Time for each individual Core Spray, LPCI, and HPCI instrument channel from 12 hours to 24 hours, it extends the time for each individual ADS channel from 12 hours to either 96 hours (if either HPCI or RCIC is inoperable) or 8 days (if both HPCI and RCIC are OPERABLE), and allows 1 hour to declare the associated ECCS subsystem inoperable when there is a loss of Function instead of requiring immediate action to place the appropriate channel in the trip condition when more than one channel is inoperable.

The purpose of CTS Table 3.2.2 Table Note 3, in part, is to allow a short period of time to restore inoperable channels before taking action. This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a transient occurring during the allowed Completion Time. The extended Completion Times (24 hours for low pressure ECCS and HPCI instrumentation channels and up to 8 days for ADS channels) are based on the reliability analyses of NEDC-30936-P-A Parts 1 and 2, December 1988, and found to be acceptable. The result of the NRC review of this generic reliability analyses as it relates to Monticello is documented in the NRC Safety Evaluation Report (SER) dated December 23, 1998. The Completion Time of one hour is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels. This change is designated as less restrictive because additional time is allowed in the ITS to restore or place the channels in trip than is allowed in the CTS.

L.4 *(Category 4 - Relaxation of Required Action)* If channels of the Core Spray and LPCI Reactor Low Pressure Permissive Bypass Timer (CTS Table 3.2.2 Function A.1.b.ii) or the ADS Functions (CTS Table 3.2.2 Functions C.1, C.2, or C.3) are inoperable and the requirements of CTS Table 3.2.2 Notes 3.(a) and 3.(b) cannot be met, then Table 3.2.2 Note 3.(c) requires Required Condition B to be taken (since this is the Required Condition listed in Table 3.2.2 for

Functions A.1.b.ii, C.1, C.2, and C.3). CTS Table 3.2.2 Required Condition B requires reactor pressure to be ≤ 150 psig. Under the same condition in the ITS, ITS 3.3.5.1 ACTION H requires the associated supported feature(s) to be immediately declared inoperable. This changes the CTS by deleting the requirement to be ≤ 150 psig and replaces it with the requirement to declare the associated feature(s) inoperable.

The purpose of CTS Table 3.2.2 Required Condition B is to place the unit in a condition where the instrumentation is not required to be OPERABLE. This change is acceptable because declaring the supported features inoperable will require entry into the appropriate ACTIONS associated with ITS 3.5.1. ITS 3.5.1 covers the system requirements for ECCS. When an ECCS instrument is inoperable declaring the supported feature inoperable is the appropriate action. ITS 3.5.1 will allow the plant to operate for a period of time for certain multiple low pressure ECCS subsystem inoperabilities (ITS 3.5.1 ACTIONS A, B, C, and D). ITS 3.5.1 also allows the plant to operate for a certain amount of time for an inoperable ADS valve (ITS 3.5.1 ACTION J) and for an inoperable ADS valve coincident with inoperable low pressure ECCS subsystems (ITS 3.5.1 ACTION K). The actions associated with CTS Table 3.2.2 do not allow this additional time therefore, this change is considered less restrictive. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.5 *(Category 10 – Changing Instrumentation Allowable Values)* CTS Table 3.2.2 and Table 3.2.8 specify the "Trip Setting" for the ECCS instrumentation Functions. The Trip Settings of CTS Table 3.2.2 Functions A.1.b.i and A.2 and Table 3.2.8 Function C.1 have been modified to reflect new Allowable Values as indicated for ITS Table 3.3.5.1-1 Functions 1.c, 1.d, 2.c, 2.d, and 3.d. In addition, two Notes have been added (Table 3.3.5.1-1 Notes (c) and (d)) to the CHANNEL CALIBRATION (ITS SR 3.3.5.1.4) associated with ITS Table 3.3.5.1-1 Functions 1.c, 1.d, 2.c, and 2.d. Note (c) states, "If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service." Note (d) states, "The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the nominal trip setpoint; otherwise, the channel shall be declared inoperable. The nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified in the Technical Requirements Manual (TRM)." In addition, the Allowable Value for ITS Table 3.3.5.1-1 Function 3.d only specifies a single Allowable Value, which is applicable for both one tank and two tank operations. This changes the CTS by requiring the ECCS instrumentation to be set consistent with the new "Allowable Value" and adds the Notes to the CHANNEL CALIBRATION (ITS SR 3.3.5.1.4) associated with ITS Table 3.3.5.1-1 Functions 1.c, 1.d, 2.c, and 2.d. The change in the term "Trip Setting" to "Allowable Value" is discussed in DOC A.8.

The purpose of the Allowable Values is to ensure the instruments function as assumed in the safety analyses. ITS Table 3.3.5.1-1 reflects Allowable Values consistent with the philosophy of General Electric ISTS, NUREG-1433. This Allowable Value has been established using the GE setpoint methodology guidance, as specified in the Monticello setpoint methodology. The analytic limits

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are derived from limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limit. The difference between the analytic limit and the Allowable Value allows for channel instrument accuracy, calibration accuracy, process measurement accuracy, and primary element accuracy. The margin between the Allowable Value and the nominal trip setpoint (NTSP) allows for instrument drift that might occur during the established surveillance period. Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift. These two verifications are statistical evaluations to provide additional assurance of the acceptability of the NTSP and may require changes to the NTSP. Use of these methods and verifications provides the assurance that if the setpoint is found conservative to the Allowable Value during surveillance testing, the instrumentation would have provided the required trip function by the time the process reached the analytic limit for the applicable events. The Notes (ITS Table 3.3.5.1-1 Notes (c) and (d)) added to the CHANNEL CALIBRATION (ITS SR 3.3.5.1.4) associated with Functions 1.c, 1.d, 2.c, and 2.d have been added at the request of the NRC, made during the review of the original ITS submittal. Furthermore, the Allowable Value for ITS Table 3.3.5.1-1 Function 3.d is acceptable for both one tank and two tank operation. Therefore, based on the above discussion, the changes to the Allowable Values are acceptable. This change is less restrictive because the less stringent Allowable Values are being applied in the ITS than were applied in the CTS.

Improved Standard Technical Specifications (ISTS) Markup and Justification for Deviations (JFDs)

CTS

3.3 INSTRUMENTATION

3.2.B, 3.2.D **3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation**

3.2.B, 3.2.D LCO 3.3.5.1 The ECCS instrumentation for each Function in Table 3.3.5.1-1 shall be OPERABLE.

3.2.B, 3.2.D APPLICABILITY: According to Table 3.3.5.1-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

	CONDITION	REQUIRED ACTION		COMPLETION TIME
Table 3.2.2 Note 3	A. One or more channels inoperable.	A.1	Enter the Condition referenced in Table 3.3.5.1-1 for the channel.	Immediately
Table 3.2.2 Notes 3.(a) and 3.(b), DOC M.2	B. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	B.1	-----NOTES----- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.a, 1.b, 2.a, ~~and~~ 2.b, 2.f, 2.h, and 2.k ----- Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	13 1 1 hour from discovery of loss of initiation capability for feature(s) in both divisions
		AND		

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION		COMPLETION TIME
Table 3.2.2 Notes 3.(a) and 3.(b), DOC M.2		B.2	------------NOTE------------ Only applicable for Functions 3.a and 3.b. ------------------------------	
			Declare High Pressure Coolant Injection (HPCI) System inoperable.	1 hour from discovery of loss of HPCI initiation capability
		AND		
		B.3	Place channel in trip.	24 hours
Table 3.2.2 Notes 3.(a) and 3.(b), Table 3.2.8 Notes 1.a and 1.b, DOC M.2	C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1	------------NOTES------------ 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.d, 1.e, 1.f, ~~1.c,~~ 2.c, 2.d, 2.e, 2.i, 2.j, 2.l, and 2.m ~~and 2.f~~. ------------------------------ 1 13	
			Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
		AND		
		C.2	Restore channel to OPERABLE status.	24 hours

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION		COMPLETION TIME
Table 3.2.8 Notes 1.a and 1.b and Required Condition C, DOC M.2	D. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	D.1	------------NOTE------------ Only applicable if HPCI pump suction is not aligned to the suppression pool. --------------------------------	
			Declare HPCI System inoperable.	1 hour from discovery of loss of HPCI initiation capability
		AND		
		D.2.1	Place channel in trip.	24 hours
		OR		
		D.2.2	Align the HPCI pump suction to the suppression pool.	24 hours
DOC M.2	E. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	E.1	------------NOTES------------ 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functio~~ns 1.d and~~ 2.g. (2) --------------------------------	
			Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for subsystems in both divisions
		AND		
		E.2	Restore channel to OPERABLE status.	7 days

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION		COMPLETION TIME
Table 3.2.2 Notes 3.(a) and 3.(b)	F. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	F.1	Declare Automatic Depressurization System (ADS) valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
		<u>AND</u>		
		F.2	Place channel in trip.	96 hours from discovery of inoperable channel concurrent with HPCI or reactor core isolation cooling (RCIC) inoperable <u>AND</u> 8 days
Table 3.2.2 Notes 3.(a) and 3.(b)	G. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	G.1	~~-----NOTE----- Only applicable for Functions 4.c, 4.e, 4.f, 4.g, 5.c, 5.e, 5.f, and 5.g.~~ Declare ADS valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
		<u>AND</u>		
		G.2	Restore channel to OPERABLE status.	96 hours from discovery of inoperable channel concurrent with HPCI or RCIC inoperable <u>AND</u> 8 days

2

CTS

ACTIONS (continued)

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
Table 3.2.2 Required Conditions A, B, and C, Table 3.2.8 Required Condition A	H. Required Action and associated Completion Time of Condition B, C, D, E, F, or G not met.	H.1 Declare associated supported feature(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTES-----

4.2 1. Refer to Table 3.3.5.1-1 to determine which SRs apply for each ECCS Function.

Table 3.2.2 Note 6, Table 3.2.8 Note 2 2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 6 hours for Functions 3.c, 3.f, ~~and 3.g~~; and (b) for up to 6 hours for Functions other than 3.c, 3.f ~~and 3.g~~ provided the associated Function or the redundant Function maintains ECCS initiation capability. and (2)

CTS	SURVEILLANCE		FREQUENCY	
Table 4.2.1	SR 3.3.5.1.1	Perform CHANNEL CHECK.	12 hours	
Table 4.2.1	SR 3.3.5.1.2	Perform CHANNEL FUNCTIONAL TEST.	[92] days	3
Table 4.2.1	SR 3.3.5.1.3	[Calibrate the trip unit.	[92] days]	3
Table 4.2.1	SR 3.3.5.1.4	[Perform CHANNEL CALIBRATION. INSERT 1	92 days]	3 12
Table 4.2.1	SR 3.3.5.1.~~5~~ 7	Perform CHANNEL CALIBRATION.	[~~18~~] 24 months	12 3
DOC M.4	SR 3.3.5.1.~~6~~ 8	Perform LOGIC SYSTEM FUNCTIONAL TEST.	[~~18~~] 24 months	12 3
	~~SR 3.3.5.1.7~~	~~Verify the ECCS RESPONSE TIME is within limits.~~	~~[18] months on a STAGGERED TEST BASIS~~	4

CTS

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INSERT 1

DOC M.2	SR 3.3.5.1.5	Perform CHANNEL FUNCTIONAL TEST.	12 months
DOC M.2	SR 3.3.5.1.6	Perform CHANNEL CALIBRATION.	12 months

Table 3.3.5.1-1 (page 1 of 6)
Emergency Core Cooling System Instrumentation

CTS

Table and Trip Function or Instrument Channel Number	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION (3)	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
	1. Core Spray System					
3.2.2 (A.1.a), 4.2.1 (ECCS 1)	a. Reactor Vessel Water Level - Low Low ~~Low, Level 1~~ (5)	1, 2, 3, 4(a), 5(a)	[4](b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] (3) SR 3.3.5.1.~~5~~ 7 (12) SR 3.3.5.1.~~6~~ 8 (12) ~~SR 3.3.5.1.7~~ (4)	≥ ~~[-113]~~ -48 inches (3)
3.2.2 (A.1.c), 4.2.1 (ECCS 2)	b. Drywell Pressure - High	1, 2, 3	[4](b)	B	~~SR 3.3.5.1.1~~ (11) SR 3.3.5.1.2 ~~[SR 3.3.5.1.3]~~ (6) SR 3.3.5.1.~~5~~ 4 SR 3.3.5.1.~~6~~ 8 (12) ~~SR 3.3.5.1.7~~ (4)	≤ ~~[1.92]~~ 2 psig (3)
3.2.2 (A.2), 4.2.1 (ECCS 4)	c. Reactor Steam Dome Pressure - Low (Injection Permissive)	1, 2, 3	~~[4]~~ 2 (15) (c)(d)	C	~~SR 3.3.5.1.1~~ (11) SR 3.3.5.1.2 ~~[SR 3.3.5.1.3]~~ (6) SR 3.3.5.1.~~5~~ 4 SR 3.3.5.1.~~6~~ 8 (12) ~~SR 3.3.5.1.7~~ (4)	≥ ~~[390]~~ 397 psig and ≤ ~~[500]~~ 440 psig (3)
		4(a), 5(a)	~~[4]~~ 2 (15) (c)(d)	B	~~SR 3.3.5.1.1~~ (11) SR 3.3.5.1.2 ~~[SR 3.3.5.1.3]~~ (6) SR 3.3.5.1.~~5~~ 4 SR 3.3.5.1.~~6~~ 8 (12) ~~SR 3.3.5.1.7~~ (4)	≥ ~~[390]~~ 397 psig and ≤ ~~[500]~~ 440 psig (3)
	INSERT 2 (1)					
	~~[d. Core Spray Pump Discharge Flow - Low (Bypass)~~ (2)	~~1, 2, 3, 4(a), 5(a)~~	~~[2] [1 per pump]~~	~~E~~	~~SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6~~	~~≥ [ ] gpm and ≤ [ ] gpm]~~
	~~[e. Manual Initiation~~	~~1, 2, 3, 4(a), 5(a)~~	~~[2] [1 per subsystem]~~	~~C~~	~~SR 3.3.5.1.6~~	~~NA]~~

(a) When associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2, "ECCS - Shutdown."

(b) Also required to initiate the associated emergency diesel generator (~~DG~~ EDG) ~~and isolate the associated plant service water (PSW) turbine building (T/B) isolation valves~~. (3)

INSERT 2A (15)

CTS

1

INSERT 2

Table and Trip Function or Instrument Channel Number							
3.2.2 (A.1.b.i), 4.2.1, (ECCS 3)	d.	Reactor Steam Dome Pressure Permissive - Low (Pump Permissive)	1, 2, 3	2	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.8 15 (c)(d)	≥ 397 psig
			4(a), 5(a)	2	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.8 15 (c)(d)	≥ 397 psig
3.2.2 (A.1.b.ii), 4.2.1 (ECCS 10)	e.	Reactor Steam Dome Pressure Permissive - Bypass Timer (Pump Permissive)	1, 2, 3	2	C	SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 18 minutes and ≤ 22 minutes
			4(a), 5(a)	2	B	SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 18 minutes and ≤ 22 minutes
DOC M.2	f.	Core Spray Pump Start - Time Delay Relay	1, 2, 3 4(a), 5(a)	1 per pump	C	SR 3.3.5.1.7 SR 3.3.5.1.8	≤ 15.86 seconds

15

INSERT 2A

(c) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(d) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the nominal trip setpoint; otherwise, the channel shall be declared inoperable. The nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified in the Technical Requirements Manual (TRM).

Table 3.3.5.1-1 (page 2 of 6)
Emergency Core Cooling System Instrumentation

CTS Table and Trip Function or Instrument Channel Number	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION (3)	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
	2. Low Pressure Coolant Injection (LPCI) System					
3.2.2 (A.1.a), 4.2.1 (ECCS 1)	a. Reactor Vessel Water Level - Low Low Low, ~~Level~~ 1 (5)	1, 2, 3, 4[(a)], 5[(a)]	[4]~~(b)~~ (7)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] (3) SR 3.3.5.1.5 7 (12) SR 3.3.5.1.6 8 ~~SR 3.3.5.1.7~~ (4)	-48 ≥ ~~[-113]~~ inches (3)
3.2.2 (A.1.c), 4.2.1 (ECCS 2)	b. Drywell Pressure - High	1, 2, 3	[4]~~(b)~~ (7)	B	(11) ~~SR 3.3.5.1.1~~ SR 3.3.5.1.2 ~~[SR 3.3.5.1.3]~~ (6) SR 3.3.5.1.5 4 SR 3.3.5.1.6 8 (12) ~~SR 3.3.5.1.7~~ (4)	2 ≤ ~~[1.92]~~ psig (3)
3.3.2 (A.2), 4.2.1 (ECCS 4)	c. Reactor Steam Dome Pressure - Low (Injection Permissive)	1, 2, 3	~~[4]~~ 2	C	(11) ~~SR 3.3.5.1.1~~ SR 3.3.5.1.2 ~~[SR 3.3.5.1.3]~~ (6) (15) (c)(d) SR 3.3.5.1.5 4 SR 3.3.5.1.6 8 (12) ~~SR 3.3.5.1.7~~ (4)	397 ≥ ~~[390]~~ psig and 440 ≤ ~~[500]~~ psig (3)
		4[(a)], 5[(a)]	~~[4]~~ 2	B	(11) ~~SR 3.3.5.1.1~~ SR 3.3.5.1.2 ~~[SR 3.3.5.1.3]~~ (6) (15) (c)(d) SR 3.3.5.1.5 4 SR 3.3.5.1.6 8 (12) ~~SR 3.3.5.1.7~~ (4)	397 ≥ ~~[390]~~ psig and 440 ≤ ~~[500]~~ psig (3)
	INSERT 3 (1)					
	~~d. Reactor Steam Dome Pressure - Low (Recirculation Discharge Valve Permissive)~~	~~1[(c)],2[(c)],3[(c)]~~	~~[4]~~	~~C~~	~~SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] SR 3.3.5.1.5 SR 3.3.5.1.6~~	~~≥ [335] psig~~ (2)
	~~e. Reactor Vessel Shroud Level - Level 0~~	~~1, 2, 3~~	~~[2]~~	~~B~~	~~SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] SR 3.3.5.1.5 SR 3.3.5.1.6~~	~~≥ [-202] inches~~ (8)

(a) When associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2~~, "ECCS - Shutdown"~~. (14)

~~(b) Also required to initiate the associated [DG and isolate the associated PSW T/B isolation valves].~~ (7)

~~(c) With associated recirculation pump discharge valve open.~~ (2)

INSERT 3A (15)

CTS

1

INSERT 3

Table and Trip Function or Instrument Channel Number						
3.2.2 (A.1.b.I), 4.2.1 (ECCS 3)	d. Reactor Steam Dome Pressure Permissive - Low (Pump Permissive)	1, 2, 3	2	C	SR 3.3.5.1.2 SR 3.3.5.1.4 15 (c)(d) SR 3.3.5.1.8	≥ 397 psig
		4[(a)], 5[(a)]	2	B	SR 3.3.5.1.2 SR 3.3.5.1.4 15 (c)(d) SR 3.3.5.1.8	≥ 397 psig
3.2.2 (A.1.b.II), 4.2.1 (ECCS 10)	e. Reactor Steam Dome Pressure Permissive - Bypass Timer (Pump Permissive)	1, 2, 3	2	C	SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 18 minutes and ≤ 22 minutes
		4[(a)], 5[(a)]	2	B	SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 18 minutes and ≤ 22 minutes

15

INSERT 3A

(c) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(d) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the nominal trip setpoint; otherwise, the channel shall be declared inoperable. The nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified in the TRM.

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

CTS

Table and Trip Function or Instrument Channel Number	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION (3)	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
	2. LPCI System			(13)		
DOC M.2	(3) [f. Low Pressure Coolant Injection Pump Start - Time Delay Relay	1, 2, 3, 4(a), 5(a)	(4) [4] [1 per pump]	(B) [D]	SR 3.3.5.1.[5] (7) SR 3.3.5.1.[6] (8) (12)	
	Pumps A,B[,D] (5)					~~≥ 9 seconds and~~ (10) ≤ [1] seconds 5.33 (3)
	Pump C (s) (,D) (5)					≤ [] second[] 10.59 (3) (s)
DOC M.2	(3) [g. Low Pressure Coolant Injection Pump Discharge ((9) Flow - Low Bypass)	1, 2, 3, 4(a), 5(a)	[4] [1 per pump]	E	(11) ~~SR 3.3.5.1.1~~ SR 3.3.5.1.2 SR 3.3.5.1.[5] (7) SR 3.3.5.1.[6] (8) (12)	≥ [] gpm and 360 (3) ≤ [] gpm [] 745
	~~[h. Manual Initiation~~	~~1, 2, 3, 4(a), 5(a)~~	~~[2] [1 per subsystem]~~	~~C~~	~~SR 3.3.5.1.6~~	~~NA]~~ (2)
	INSERT 4 (1)					
	3. High Pressure Coolant Injection (HPCI) System					
3.2.2 (B.2), 4.2.1 (ECCS 1)	a. Reactor Vessel Water Level - Low Low, ~~Level 2~~ (5)	1, 2[(d)], 3[(d)] (e) (2)	[4]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] (3) SR 3.3.5.1.[5] (7) SR 3.3.5.1.[6] (8) (12) ~~SR 3.3.5.1.7~~ (4)	≥ [-47] inches -48
3.2.2 (B.1), 4.2.1 (ECCS 2)	b. Drywell Pressure – High	1, 2[(d)], 3[(d)] (e) (2)	[4]	B	(11) ~~SR 3.3.5.1.1~~ SR 3.3.5.1.2 ~~[SR 3.3.5.1.3]~~ (6) SR 3.3.5.1.[5] (4) SR 3.3.5.1.[6] (8) (12) ~~SR 3.3.5.1.7~~ (4)	≤ [1.92] psig 2 (3)
3.2.8 (B.1), 4.2.1 (ECCS 9)	c. Reactor Vessel Water Level - High[, ~~Level 8~~] (5)	1, 2[(d)], 3[(d)] (e) (2)	[2]	C	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] (3) SR 3.3.5.1.[5] (7) SR 3.3.5.1.[6] (8) (12) ~~SR 3.3.5.1.7~~ (4)	≤ [56.5] inches 48 (3)

3.2.2 Function D (a) When associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2, ~~"ECCS - Shutdown."~~ (14)

~~[(d)]~~ (e) With reactor steam dome pressure > [150] psig. (2) (3)

CTS

1

INSERT 4

DOC M.2	h.	Reactor Steam Dome Pressure - Low (Break Detection)	1, 2, 3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 873.6 psig and ≤ 923.4 psig
DOC M.2	i.	Recirculation Pump Differential Pressure - High (Break Detection)	1, 2, 3	4 per pump	C	SR 3.3.5.1.2 SR 3.3.5.1.7 SR 3.3.5.1.8	≤ 63.5 inches wc
DOC M.2	j.	Recirculation Riser Differential Pressure - High (Break Detection)	1, 2, 3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.6 SR 3.3.5.1.8	≤ 24.0 inches wc
DOC M.2	k.	Reactor Steam Dome Pressure - Time Delay Relay (Break Detection)	1, 2, 3	2	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.8	≤ 2.79 seconds
DOC M.2	l.	Recirculation Pump Differential Pressure - Time Delay Relay (Break Detection)	1, 2, 3	2	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.8	≤ 0.716 seconds
DOC M.2	m.	Recirculation Riser Differential Pressure - Time Delay Relay (Break Detection)	1, 2, 3	2	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.8	≤ 0.697 seconds

CTS

Table 3.3.5.1-1 (page 4 of 6)
Emergency Core Cooling System Instrumentation

Table and Trip Function or Instrument Channel Number	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
	3. HPCI System					
3.2.8 (C.1), 4.2.1 (ECCS 8)	d. Condensate Storage Tank Level - Low	1, 2[(f)], 3[(f)] (e)	[2]	D	~~[SR 3.3.5.1.1]~~ ~~SR 3.3.5.1.2~~ ~~[SR 3.3.5.1.4]~~ SR 3.3.5.1.7 SR 3.3.5.1.[6] 8	≥ [0] 29.3 inches
DOC M.2	e. Suppression Pool Water Level - High	1, 2[(f)], 3[(f)] (e)	[2]	D	~~SR 3.3.5.1.1~~ SR 3.3.5.1.[2] 5 ~~[SR 3.3.5.1.3]~~ SR 3.3.5.1.[5] 6 SR 3.3.5.1.[6] 8	≤ [154] 3.0 inches
DOC M.2	[f. High Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)	1, 2[(f)], 3[(f)] (e)	[1]	E	~~SR 3.3.5.1.1~~ 5 SR 3.3.5.1.[2] SR 3.3.5.1.[5] 6 SR 3.3.5.1.[6] 8	≥ [ ] 362 gpm and ≤ [ ] 849 gpm]
	~~[g. Manual Initiation~~	~~1, 2[(d)], 3[(d)]~~	~~[1]~~	~~C~~	~~SR 3.3.5.1.6~~	~~NA]~~
	4. Automatic Depressurization System (ADS) Trip System A					
3.2.2 (C.1), 4.2.1 (ECCS 1)	a. Reactor Vessel Water Level - Low Low ~~Low, Level 1~~	1, 2[(f)], 3[(f)] (e)	[2]	F	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] SR 3.3.5.1.[5] 7 SR 3.3.5.1.[6] 8	≥ [-1/13] -48 inches
	~~b. Drywell Pressure - High~~	~~1, 2[(d)], 3[(d)]~~	~~[2]~~	~~F~~	~~SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] SR 3.3.5.1.5 SR 3.3.5.1.6~~	~~≤ [1.92] psig~~
3.2.2 (C.2), 4.2.1 (ECCS 11)	[c] b. Automatic Depressurization System Initiation Timer	1, 2[(f)], 3[(f)] (e)	[1]	G	[SR 3.3.5.1.[5] 7 SR 3.3.5.1.[6] 8	≤ [120] seconds

[(f)] (e) With reactor steam dome pressure > [150] psig.

CTS

Table 3.3.5.1-1 (page 5 of 6)
Emergency Core Cooling System Instrumentation

Table and Trip Function or Instrument Channel Number	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
	4. ADS Trip System A					
	d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory)	1, 2(d), 3(d)	[1]	F	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] SR 3.3.5.1.5 SR 3.3.5.1.6	≥ [10] inches
3.2.2 (C.3), 4.2.1 (ECCS 6)	e c. Core Spray Pump Discharge Pressure - High	1, 2(e), 3(e)	[2]	G	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] SR 3.3.5.1.5 4 SR 3.3.5.1.6 8	≥ [137] 75 psig and ≤ [ ] 125 psig
3.2.2 (C.3), 4.2.1 (ECCS 6)	f d. Low Pressure Coolant Injection Pump Discharge Pressure - High	1, 2(e), 3(e)	[4]	G	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] SR 3.3.5.1.5 4 SR 3.3.5.1.6 8	≥ [112] 75 psig and ≤ [ ] 125 psig
	g. Automatic Depressurization System Low Water Level Actuation Timer	1, 2(d), 3(d)	[2]	G	[SR 3.3.5.1.5] SR 3.3.5.1.6	≤ [13] minutes
	[h. Manual Initiation	1, 2(d), 3(d)	[2]	G	SR 3.3.5.1.6	N/A]
	5. ADS Trip System B					
3.2.2 (C.1), 4.2.1 (ECCS 1)	a. Reactor Vessel Water Level - Low Low Low, Level 1	1, 2(e), 3(e)	[2]	F	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] SR 3.3.5.1.5 7 SR 3.3.5.1.6 8	≥ [-113] -48 inches
	b. Drywell Pressure - High	1, 2(d), 3(d)	[2]	F	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] SR 3.3.5.1.5 SR 3.3.5.1.6	≤ [1.92] psig
3.2.2 (C.2) 4.2.1 (ECCS 11)	c b. Automatic Depressurization System Initiation Timer	1, 2(e), 3(e)	[1]	G	[SR 3.3.5.1.5] 7 SR 3.3.5.1.6 8	≤ [120] seconds

(d) (e) With reactor steam dome pressure > [150] psig.

INSERT 5

2 3 4 6 7 8 11 12 15 (c)(d)

15

INSERT 5

(c) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(d) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the nominal trip setpoint; otherwise, the channel shall be declared inoperable. The nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified in the TRM.

CTS

Table 3.3.5.1-1 (page 6 of 6)
Emergency Core Cooling System Instrumentation

Table and Trip Function or Instrument Channel Number	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
	5. ADS Trip System B					
	d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory)	1, 2(d), 3(d)	[1]	F	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] SR 3.3.5.1.5 SR 3.3.5.1.6	≥ [10] inches
3.2.2 (C.3), 4.2.1 (ECCS 6)	c e. Core Spray Pump Discharge Pressure - High	1, 2(e), 3(e)	[2]	G	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] SR 3.3.5.1.5 4 SR 3.3.5.1.6 8 (c)(d)	≥ [137] 75 psig and ≤ [7] 125 psig
3.2.2 (C.3), 4.2.1 (ECCS 6)	d f. Low Pressure Coolant Injection Pump Discharge Pressure - High	1, 2(e), 3(e)	[4]	G	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] SR 3.3.5.1.5 4 SR 3.3.5.1.6 8 (c)(d)	≥ [112] 75 psig and ≤ [7] 125 psig
	g. Automatic Depressurization System Low Water Level Actuation Timer	1, 2(d), 3(d)	[2]	G	[SR 3.3.5.1.5] SR 3.3.5.1.6	≥ [13] minutes
	[h. Manual Initiation	1, 2(d), 3(d)	[2]	G	SR 3.3.5.1.6	NA]

(d) (e) With reactor steam dome pressure > [150] psig.

INSERT 6

15 **INSERT 6**

(c) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(d) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the nominal trip setpoint; otherwise, the channel shall be declared inoperable. The nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified in the TRM.

Attachment 1, Volume 8, Rev. 1, Page 360 of 763

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

1. Eleven ECCS Functions have been added to ISTS Table 3.3.5.1-1. ITS Function 1.d, Reactor Steam Dome Pressure Permissive - Low (Pump Permissive), 1.e, Reactor Steam Dome Pressure Permissive - Bypass Timer, and Function 1.f, Core Spray Pump Start -Time Delay Relay, are associated with the Core Spray System. ITS Table 3.3.5.1-1 Functions 1.d and 1.e have been added consistent with the CTS. ITS Table 3.3.5.1-1 Function 1.f is similar to ISTS Table 3.3.5.1-1 Function 2.f. ITS Function 2.d, Reactor Steam Dome Pressure Permissive - Low (Pump Permissive) and Function 2.e, Reactor Steam Dome Pressure Pump Permissive - Bypass Timer, are associated with the LPCI System and are consistent with the CTS. The other Functions have been added to ensure the Loop Select Logic of the LPCI System functions properly. These Functions are ITS Functions 2.h, 2.i, 2.j, 2.k, 2.l, and 2.m. Since these Functions have been added, Note 2 to Required Action B.1 and Note 2 to Required Action C.1 have been revised.

2. ISTS Table 3.3.5.1-1 Functions 1.d, 1.e, 2.d, 2.h, 3.g, 4.b, 4.d, 4.g, 4.h, 5.b, 5.d, 5.g, and 5.h have not been included in the Monticello ITS since they are not consistent with the plant specific design or licensing basis. Subsequent Functions, have been renumbered as required. Footnote (c) to ITS Table 3.3.5.1-1 has been deleted since Function 2.d has been deleted. Subsequent Footnotes have been renumbered as necessary. In addition, the ISTS 3.3.5.1 Required Action G.1 Note has been deleted since the Required Action now applies to each Function that references Condition G. Furthermore, Surveillance Requirements Note 2 has been modified due to the deletion of ISTS Table 3.3.5.1-1 Function 3.g.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. The ECCS RESPONSE TIME requirement (ISTS SR 3.3.5.1.7) has not been adopted, consistent with Monticello current licensing basis. Monticello response time requirements reflect the industry standards and regulations to which the plant has been committed to and licensed to since the operating license was granted. Monticello is committed to the testing requirements contained in IEEE-279-1968 and IEEE-338-1971. These industry standards provide guidance and requirements for conducting periodic testing of protection systems. IEEE-279-1968 does not address response time testing. Response time testing requirements do not appear in IEEE-338 until the 1975 revision.

5. The proper Monticello plant specific nomenclature/value/design requirements have been provided.

6. The Surveillance Requirements associated with specific Functions in ISTS Table 3.3.5.1-1 have been revised to be consistent with the current licensing basis or with the setpoint calculation methodology.

7. The LPCI Initiation signals do not start the emergency diesel generators. Therefore, Footnote (b) does not apply to ITS Function 2.a or 2.b, and reference to it has been deleted.

8. ISTS Table 3.3.5.1-1 Function 2.e, Reactor Vessel Shroud Level - Level 0, has been not been included to the Monticello ITS since it does not satisfy one of the four

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

criteria established in 10 CFR 50.36(c)(2)(ii). This is also consistent with the current licensing basis.

9. Grammatical/editorial error corrected.

10. ISTS Table 3.3.5.1-1 Function 2.f requires a minimum time for the ECCS pump start time delay relays. The ISTS Bases states that the minimum time is to ensure that excess loading will not cause failure of the power source; i.e., the minimum Allowable Value is chosen to be long enough so that most of the starting transient of the first pump is complete before starting the second pump on the same 4.16 kV emergency bus. Failure of this portion of the instrumentation will result in the EDG or offsite circuit being inoperable; it does not necessarily result in the inoperability of the ECCS pump. The ECCS analysis assumes the pumps are operating at a certain time; starting the pumps sooner than assumed does not invalidate the ECCS analysis. This requirement is adequately covered by ISTS SR 3.8.1.19 (ITS SR 3.8.1.12), which requires the verification of proper operation of the EDG following an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ECCS initiation signal. This test confirms that the loads are properly loaded onto the EDG and that the EDG can maintain adequate voltage and frequency when the associated loads are sequenced onto the EDG. Therefore, if a time delay relay actuated too soon such that a power source was affected, the requirements of ITS SR 3.8.1.12 would not be met and the affected EDG or offsite circuit would be declared inoperable and the ACTIONS of ITS 3.8.1 taken. Therefore, there is no reason to require minimum times in the ECCS Instrumentation Specification. This is also consistent with current licensing basis, which does not have minimum time requirements for the ECCS pump start time delay relays in the ECCS Instrumentation Specification.

11. The CHANNEL CHECK Surveillance in ISTS Table 3.3.5.1-1 associated with these Functions (ISTS Table 3.3.5.1-1 Functions 1.b, 1.c, 2.b, 2.c, 2.g, 3.b, 3.d, 3.e, 3.f, 4.e, 4.f, 5.e, and 5.f) have been deleted because the associated channels at Monticello include only a switch and there is no available method to verify channel indication.

12. Two new Surveillance Requirments have been added (SR 3.3.5.1.5, a CHANNEL FUNCTIONAL TEST, and SR 3.3.5.1.6, a CHANNEL CALIBRATION) at a 12 month Frequency to ensure ITS Table 3.3.5.1-1 Functions 3.e and 3.f remain OPERABLE. The proposed Surveillance Frequencies are consistent with current practice and setpoint methodology. Subsequent SRs have been renumbered.

13. ISTS Table 3.3.5.1-1 Function 2.f, Low Pressure Coolant Injection Pump Start - Time Delay Relay, requires entry into ITS 3.3.5.1 ACTION C if one or more channels are inoperable. ITS 3.3.5.1 Required Action C.2 requires the channel to be restored to OPERABLE status within 24 hours; it does not allow the channel to be placed in trip and continue to operate. For ISTS Table 3.3.5.1-1 Function 2.f, this action (restore the channel) is required because the standard design only includes one channel for each LPCI pump, and placing the channel in trip will result in an immediate initiation without time delay when an initiation signal is received. At Monticello, the Low Pressure Coolant Injection Pump Start - Time Delay Relay Function for each LPCI pump is developed by four time delay relays. Each time delay relay will start when there is a LOCA signal present and power is available on the associated 4.16 kV

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

essential bus. After the time delay relay times out, a signal is sent to start the associated LPCI pump. The outputs of the time delay relays are arranged in a one-out-of-two-taken-twice logic for each LPCI pump. Therefore, based on the trip logic, if a channel was placed in trip after the 24 hour Completion Time, this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Furthermore, placing the channel in trip would not result in an immediate initiation without time dely when an initiation signal is received. Therefore, ITS Table 3.3.5.1-1 Function 2.f has been modified to require entry into ITS 3.3.5.1 ACTION B instead of ACTION C. ITS 3.3.5.1 Required Action B.3 requires the inoperable channel to be placed in trip within 24 hours. In addition, Function 2.f has been deleted from ISTS 3.3.5.1 Required Action C.1 Note 2 and added to ITS 3.3.5.1 Required Action B.1 Note 2.

14. The Title of the LCO has already been provided on the first page of the Table. Therefore, it is not required to repeat the title in each subsequent use of the LCO number.

15. These Notes associated with SR 3.3.5.1.4 for Functions 1.c, 1.d, 2.c, 2.d, 4.c, 4.d, 5.c, and 5.d have been added in accordance with the RAI 200510281246 and RAI 200510281248.

Improved Standard Technical Specifications (ISTS) Bases Markup and Justification for Deviations (JFDs)

B 3.3 INSTRUMENTATION

B 3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

BASES

BACKGROUND

The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient.

For most anticipated operational occurrences and Design Basis Accidents (DBAs), a wide range of dependent and independent parameters are monitored.

The ECCS instrumentation actuates core spray (CS), low pressure coolant injection (LPCI), high pressure coolant injection (HPCI), E Automatic Depressurization System (ADS), and the emergency diesel generators (DGs). The equipment involved with each of these systems is described in the Bases for LCO 3.5.1, "ECCS - Operating" and LCO 3.8.1, "AC Sources - Operating." 1 2

Core Spray System

The CS System may be initiated by either automatic or manual means, although manual initiation requires manipulation of individual pump and valve control switches. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low ~~Low, Level 1~~ or Drywell Pressure - High. INSERT 1 ~~Each of these diverse variables is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of the eight trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic (i.e., two trip systems) for each Function.~~ INSERT 1A 1 1 1

~~The high drywell pressure initiation signal is a sealed in signal and must be manually reset. The CS System can be reset if reactor water level has been restored, even if the high drywell pressure condition persists. The logic can also be initiated by use of a manual push button (one push button per subsystem).~~ Upon receipt of an initiation signal, the CS pumps are started ~~immediately~~ in approximately 15 seconds after AC power is available. INSERT 1AA 1 1

The CS test line isolation valve, which is also a primary containment isolation valve (PCIV), is closed on a CS initiation signal to allow full system flow assumed in the accident analyses and maintain primary containment isolated in the event CS is not operating.

~~The CS pump discharge flow is monitored by a flow transmitter. When the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow assumed in the accident analysis.~~ 1

1 **INSERT 1**

The Reactor Vessel Water Level - Low Low initiation signal is generated coincident with Reactor Steam Dome Pressure - Low (Pump Permissive) or if the Reactor Vessel Water Level - Low Low signal is sustained for approximately 20 minutes.

1 **INSERT 1A**

The Reactor Vessel Water Level - Low Low variable is monitored by four redundant transmitters, connected to four trip units. The outputs of the four trip units are connected to relays whose contacts are directed to two trip systems and the logic in each trip system is arranged in a one-out-of-two taken twice logic. The Drywell Pressure - High variable is monitored by four redundant pressure switches. The outputs of the switches are connected to relays whose contacts are directed to two trip systems and the logic in each trip system is arranged in a one-out-of-two taken twice logic. The Reactor Steam Dome Pressure - Low (Pump Permissive) variable is monitored by two redundant switches. The outputs of the switches are connected to relays whose contacts are directed to two trip systems and the logic in each trip system is arranged in a one-out-of-two logic. Each trip system will delay CS pump start and valve logic on low low reactor vessel water level until reactor steam dome pressure has fallen to a value below the CS System's maximum design pressure. The Reactor Steam Dome Pressure Permissive - Bypass Timer (Pump Permissive) variable is developed by two redundant time delay relays. A time delay relay is located in each trip system and a contact associated with this relay (one-out-of-one logic for each trip system) will bypass the Reactor Steam Dome Pressure - Low (Pump Permissive) after the timer has timed out. The CS pumps start and valve logic will receive the high drywell pressure signals without delay. The Reactor Steam Dome Pressure - Low (Injection Permissive) variable is monitored by two redundant pressure switches. The outputs of the switches are connected to relays whose contacts input into two trip systems. Each trip system is arranged in a one-out-of-two logic. Each trip system will delay CS injection valve actuation logic until reactor steam dome pressure has fallen to a value below the CS System's maximum design pressure regardless of the initiation signal. Each trip system will open the associated CS subsystem valves.

1 **INSERT 1AA**

The Core Spray Pump Start - Time Delay Relay Function for each CS pump is developed by one time delay relay. The time delay relay starts when there is a LOCA signal present and power is available on the associated 4.16 kV essential bus. After the time delay relay times out, the associated CS pump starts.

BASES

BACKGROUND (continued)

~~The CS System also monitors the pressure in the reactor to ensure that, before the injection valves open, the reactor pressure has fallen to a value below the CS System's maximum design pressure. The variable is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.~~ 1

Low Pressure Coolant Injection System

, although manual initiation requires manipulation of individual pump and valve control switches

The LPCI is an operating mode of the Residual Heat Removal (RHR) System, with two LPCI subsystems. The LPCI subsystems may be initiated by automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low ~~Low, Level 1~~, Drywell Pressure - High, or both. ~~Each of these diverse variables is monitored by four redundant transmitters which, in turn, are connected to four trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic (i.e., two trip systems) for each Function. Once an initiation signal is received by the LPCI control circuitry, the signal is sealed in until manually reset.~~ 1

INSERT 2

INSERT 2A 1

Upon receipt of an initiation signal, the LPCI ~~C~~ pump s starts after a ~~0.5 second delay when power is available. The LPCI A, B, and D pumps are started after a 10 second delay to limit the loading of the standby power sources.~~ 1

INSERT 3 1

Each LPCI subsystem's discharge flow is monitored by a flow ~~transmitter~~ switch. When a pump is running and discharge flow is low enough so that pump overheating may occur, the respective minimum flow return line valve is opened after an approximate 10 second time delay. If flow is above the minimum flow setpoint, the valve is automatically closed to allow the full system flow assumed in the analyses. 1 1

The RHR test line suppression pool cooling isolation valve, suppression pool spray isolation valves, and containment spray isolation valves (which are also PCIVs) are also closed on a LPCI initiation signal to allow the full system flow assumed in the accident analyses and maintain primary containment isolated in the event LPCI is not operating.

1 **INSERT 2**

The Reactor Vessel Water Level - Low Low initiation signal is generated coincident with Reactor Steam Dome Pressure - Low (Pump Permissive) or if the Reactor Vessel Water Level - Low Low signal is sustained for approximately 20 minutes.

1 **INSERT 2A**

The Reactor Vessel Water Level - Low Low variable is monitored by four redundant transmitters, connected to four trip units. The outputs of the four trip units are connected to relays whose contacts are directed to two trip systems and the logic in each trip system is arranged in a one-out-of-two taken twice logic. The Drywell Pressure - High variable is monitored by four redundant pressure switches. The outputs of the switches are connected to relays whose contacts are directed to two trip systems and the logic in each trip system is arranged in a one-out-of-two taken twice logic. The Reactor Steam Dome Pressure - Low (Pump Permissive) variable is monitored by two redundant switches. The outputs of the switches are connected to relays whose contacts are directed to two trip systems and the logic in each trip system is arranged in a one-out-of-two logic. Each trip system will delay LPCI pump start and valve logic on low low reactor vessel water level until reactor steam dome pressure has fallen to a value below the LPCI System's maximum design pressure. The Reactor Steam Dome Pressure Permissive - Bypass Timer (Pump Permissive) variable is developed by two redundant time delay relays. A time delay relay is located in each trip system and a contact associated with this relay (one-out-of-one logic for each trip system) will bypass the Reactor Steam Dome Pressure - Low (Pump Permissive) after the timer has timed out. The LPCI pumps start and valve logic will receive the high drywell pressure signals without delay. The Reactor Steam Dome Pressure - Low (Injection Permissive) variable is monitored by two redundant pressure switches. The outputs of the switches are connected to relays whose contacts input into two trip systems. Each trip system is arranged in a one-out-of-two logic. Each trip system will delay LPCI injection valve actuation logic until reactor steam dome pressure has fallen to a value below the LPCI System's maximum design pressure regardless of the initiation signal. Each trip system will open the associated LPCI subsystem valves.

1 **INSERT 3**

are automatically started (pumps A and B approximately 5 seconds after AC power is available and pumps C and D approximately 10 seconds after AC power is available). The Low Pressure Coolant Injection Pump Start - Time Delay Relay Function for each LPCI pump is developed by four time delay relays. Each time delay relay will start when there is a LOCA signal present and power is available on the associated 4.16 kV essential bus. After a time delay relay times out, a signal is sent to start the associated LPCI pump. The outputs of the time delay relays are arranged in a one-out-of-two taken twice logic for each LPCI pump.

BASES

BACKGROUND (continued)

INSERT 4

~~The LPCI System monitors the pressure in the reactor to ensure that before an injection valve opens, the reactor pressure has fallen to a value below the LPCI System's maximum design pressure. The variable is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic. Additionally, instruments are provided to close the recirculation pump discharge valves to ensure that LPCI flow does not bypass the core when it injects into the recirculation lines. The variable is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.~~ 1

~~Low reactor water level in the shroud is detected by two additional instruments to automatically isolate other modes of RHR (e.g., suppression pool cooling) when LPCI is required. Manual overrides for these isolations are provided.~~ 3

High Pressure Coolant Injection System

The Reactor Vessel Water Level - Low Low

The HPCI System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low ~~Level 2~~ or Drywell Pressure - High. ~~Each of these~~ variable~~s~~ is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic ~~for each Function~~. 3 1 1

The Drywell Pressure - High variable is monitored by four switches. The outputs of the switches are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.

switch

The HPCI pump discharge flow is monitored by a flow ~~transmitter~~. When ~~the pump is running~~ and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow assumed in the accident analysis. 1 1

an initiation signal is present (Reactor Vessel Water Level - Low Low or Drywell Pressure - High)

return s to the condensate storage tanks (CSTs) are

The HPCI test line ~~isolation~~ valve ~~(which is also a PCIV) is~~ closed upon receipt of a HPCI initiation signal to allow the full system flow assumed in the accident analysis ~~and maintain primary containment isolated in the event HPCI is not operating~~. 1

1

INSERT 4

The LPCI System initiation logic also contains LPCI Loop Select Logic whose purpose is to identify and direct LPCI flow to the unbroken recirculation loop if a Design Basis Accident (DBA) occurs. The LPCI Loop Select Logic is initiated upon the receipt of either a Reactor Vessel Water Level - Low Low signal or a Drywell Pressure - High signal, as discussed previously. When initiated, the LPCI Loop Select Logic first determines recirculation pump operation by sensing the differential pressure (dp) between the suction and discharge of each pump. There are four dp switches monitoring each recirculation loop that are, in turn, connected to relays whose contacts are connected to two trip systems. The dp switches will trip when the dp across the pump is greater than a predetermined value. The contacts are arranged in a one-out-of-two taken twice logic for each recirculation pump. If the logic senses that either pump is not running, i.e., single loop operation, then a trip signal is sent to both recirculation pumps to eliminate the possibility of pipe breaks being masked by the operating recirculation pump pressure. However, the pump trip signal is delayed approximately 0.5 seconds to ensure that at least one pump is off since the break detection sensitivity is greater with both pumps running. If a pump trip signal is generated, reactor steam dome pressure must drop to a specified value before the logic will continue. This adjusts the selection time to optimize sensitivity and still ensure that LPCI injection is not unnecessarily delayed. The reactor steam dome pressure is sensed by four pressure switches that are, in turn, connected to relays whose contacts are connected to two trip systems. The contacts are arranged in a one-out-of-two taken twice logic. After the satisfaction of this pressure requirement or if both recirculation pumps indicate they are running, a 2 second time delay is provided to allow momentum effects to establish the maximum differential pressure for loop selection. Selection of the unbroken recirculation loop is then initiated. This is done by comparing the absolute pressure of the two recirculation riser loops. The broken loop is indicated by a lower pressure than the unbroken loop. The loop with the higher pressure is then used for LPCI injection. If, after a small time delay (approximately 0.5 seconds), the pressure in loop A is not indicating higher than loop B, the logic will provide a signal to close the B recirculation loop discharge valve, open the LPCI injection valve to the B recirculation loop and close the LPCI injection valve to the A recirculation loop. This is the "default" choice in the LPCI Loop Select Logic. If recirculation loop A pressure indicates higher than loop B pressure (> 1 psig), the recirculation discharge valve in loop A is closed, the LPCI injection valve to loop A is signaled to open and the LPCI injection valve to loop B is signaled to close. The four dp switches that provide input to this portion of the logic detect the pressure difference between the corresponding risers to the jet pumps in each recirculation loop. The four dp switches are connected to relays whose contacts are connected to two trip systems. The contacts in each trip system are arranged in a one-out-of-two taken twice logic. There are two redundant trip systems in the LPCI Loop Select Logic. The complete logic in each trip system must actuate for operation of the LPCI Loop Select Logic.

BASES

BACKGROUND (continued)

two

The HPCI System also monitors the water levels in the ~~condensate storage tank (CST)~~ and the suspension pool because these are the two sources of water for HPCI operation. Reactor grade water in the CST is the normal source. Upon receipt of a HPCI initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position) unless both suppression pool suction valves are open. If the water level in ~~the~~ CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. Two level switches are used to detect low water level in the CST. Either switch can cause the suppression pool suction valves to open and the CST suction valve to close. The suppression pool suction valves also automatically open and the CST suction valve closes if high water level is detected in the suppression pool. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

s s any (one on each CST) (one-out-of-two logic)

1 1 1 1

System

The HPCI provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level - High ~~Level 8~~ trip, at which time the HPCI turbine trips, which causes the turbine's stop valve ~~and the injection valves~~ to close. The logic is two-out-of-two to provide high reliability of the HPCI System. The HPCI System automatically restarts if a Reactor Vessel Water Level - Low Low ~~, Level 2~~ signal is subsequently received.

2 3 3

Automatic Depressurization System

, although manual initiation requires manipulation of each individual ADS valve control switch

The ADS may be initiated by either automatic or manual means. Automatic initiation occurs when signals indicating Reactor Vessel Water Level - Low Low ~~Low, Level 1~~; Drywell Pressure - High or ~~ADS Bypass Low Water Level Actuation Timer; confirmed Reactor Vessel Water Level - Low, Level 3~~; and CS or LPCI Pump Discharge Pressure - High are all present and the ADS Initiation Timer has timed out. There are two transmitters ~~each for~~ Reactor Vessel Water Level - Low Low ~~Low, Level 1~~ and Drywell Pressure - High, and ~~one transmitter for confirmed Reactor Vessel Water Level - Low, Level 3~~ in each of the two ADS trip systems. Each of these transmitters connects to a trip unit, which then drives a relay whose contacts form the initiation logic.

that monitor

1 1 1

BASES

BACKGROUND (continued)

to be

is

Each ADS trip system includes a time delay between satisfying the initiation logic and the actuation of the ADS valves. The ADS Initiation Timer time delay setpoint chosen ~~is~~ long enough that the HPCI has Low Low (4) sufficient operating time to recover to a level above ~~Level 1~~, yet not so (3) long that the LPCI and CS Systems are unable to adequately cool the fuel if the HPCI fails to maintain that level. An alarm in the control room is annunciated when either of the timers is timing. Resetting the ADS initiation signals resets the ADS Initiation Timers.

However, only the switches from the pumps in the associated division are required to be OPERABLE for each trip system

switches all

The ADS also monitors the discharge pressures of the four LPCI pumps and the two CS pumps. Each ADS trip system includes two discharge pressure permissive ~~transmitters~~ from ~~both~~ CS and ~~from two~~ LPCI pumps ~~in the associated Division~~ (i.e., Division 1 LPCI subsystems A and ~~D~~ input to ADS trip system A, and Division 2 LPCI subsystems B and ~~C~~ input to ADS trip system B). The signals are used as a permissive for ADS actuation, indicating that there is a source of core coolant available once the ADS has depressurized the vessel. Any one of the six low pressure pumps is sufficient to permit automatic depressurization. (1)

CS A and C

CS B and D

. Each string also has a contact that represents a CS or LPCI pump discharge pressure signal.

The ADS logic in each trip system is arranged in two strings. Each string has a contact from ~~each of the following variables:~~ Reactor Vessel Water Level - Low Low ~~Low, Level 1; Drywell Pressure - High; or Low Water Level Actuation Timer. One of the two strings in each trip system must also have a confirmed Reactor Vessel Water Level - Low, Level 3.~~ All contacts in both logic strings must close, the ADS initiation timer must time out ~~, and a CS or LPCI pump discharge pressure signal must be present~~ to initiate an ADS trip system. Either the A or B trip system will cause all the ADS relief valves to open. ~~Once the Drywell Pressure - High signal, the ADS Low Water Level Actuation Timer, or the ADS initiation signal is present, it is individually sealed in until manually reset.~~ (1)

and

INSERT 5

Manual inhibit switches are provided in the control room for the ADS; however, their function is not required for ADS OPERABILITY (provided ADS is not inhibited when required to be OPERABLE).

Emergency

E Diesel Generators (EDGs) (1)

The E DGs may be initiated by either automatic or manual means. (1)
Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low ~~Low, Level 1~~ or Drywell Pressure - High. The E DGs are also (1)
initiated upon loss of voltage signals. (Refer to the Bases for LCO 3.3.8.1, "Loss of Power (LOP) Instrumentation," for a discussion of these signals.) ~~Each of these diverse~~ variable~~s~~ is monitored by four (1)

The Reactor Vessel Water Level - Low Low

1 **INSERT 5**

The Reactor Vessel Water Level - Low Low signal in one string will seal in once both the Reactor Vessel Water Level - Low Low signal and the associated CS or LPCI pump discharge pressure signal is present. The other Reactor Vessel Water Level - Low Low signal in the other string will seal in once both the Reactor Vessel Water Level - Low Low signal and the associated CS or LPCI pump discharge pressure signal is present and the ADS Initiation timer has timed out. The signals can be manually reset.

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BASES

BACKGROUND (continued)

INSERT 5A

redundant transmitters, which are, in turn, connected to four trip units. both The outputs of the four trip units are connected to relays whose contacts are ~~connected to~~ a one-out-of-two taken twice logic to initiate ~~all three~~ E DGs ~~(2A, 1B, and 2C)~~. INSERT 6 The DGs receive their initiation signals from the CS System initiation logic. The DGs can also be started manually from the control room and locally from the associated DG room. The DG initiation signal is a ~~sealed~~ in signal and must be ~~manually~~ reset. The DG initiation logic ~~is reset~~ by resetting the associated ECCS initiation logic. E Upon receipt of a loss of coolant accident (LOCA) initiation signal, each DG is automatically started, is ready to load in approximately ~~12~~ 10 seconds, E and will run in standby conditions (rated voltage and speed, with the DG output breaker open). The DGs will only energize their respective 4.16 kV essential ~~Engineered Safety Feature~~ buses if a loss of offsite power occurs. (Refer to Bases for LCO 3.3.8.1.) ① ④

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The actions of the ECCS are explicitly assumed in the safety analyses of References 1, 2, and 3. The ECCS is initiated to preserve the integrity of the fuel cladding by limiting the post LOCA peak cladding temperature to less than the 10 CFR 50.46 limits. ①

ECCS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the ECCS instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.5.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Each ECCS subsystem ~~must~~ also respond within its ~~assumed~~ response time. ③ Table 3.3.5.1-1 is modified by two footnotes. Footnote (a) is added to clarify that the associated functions are required to be OPERABLE in MODES 4 and 5 only when their supported ECCS are required to be OPERABLE per LCO 3.5.2, ECCS - Shutdown. Footnote (b) is added to E show that certain ECCS instrumentation Functions also perform DG initiation and actuation of other ~~Technical~~ Specifications (TS) equipment. ① ③

Allowable Values are specified for each ECCS Function specified in the table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. ④

1 **INSERT 5A**

directed to two trip systems and the logic in each trip system is arranged in

4 **INSERT 6**

The Drywell Pressure - High variable is monitored by four switches. The outputs of the switches are connected to relays whose contacts are directed to two trip systems and the logic in each trip system is arranged in a one-out-of-two taken twice logic to initiate both EDGs.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. ~~The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined, accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.~~ INSERT 6A 1

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions that may require ECCS (or DG) initiation to mitigate the consequences of a design basis transient or accident. To ensure reliable ECCS and DG function, a combination of Functions is required to provide primary and secondary initiation signals. E 1 1

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Core Spray and Low Pressure Coolant Injection Systems

1.a, 2.a. Reactor Vessel Water Level - Low Low ~~Low, Level 1~~ 3

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The low pressure ECCS and associated E DGs are initiated at ~~Level 1~~ Low Low to ensure that core spray and flooding functions are available to prevent or minimize fuel damage. The Reactor Vessel Water Level - Low Low ~~Low, Level 1~~ is one of the Functions assumed to be OPERABLE and capable of initiating the ECCS during the transients analyzed in ~~References 1 and 3~~ 2. In addition, the Reactor Vessel Water Level - Low Low ~~Low, Level 1~~ Function is directly assumed in the analysis of the recirculation line break (Ref. ~~2~~ 1). The core cooling function of the ECCS, along with the scram action of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. 1 3 3 1 3 1

1

INSERT 6A

The Allowable Values and nominal trip setpoints (NTSP) are derived, using the General Electric setpoint methodology guidance, as specified in the Monticello setpoint methodology. The Allowable Values are derived from the analytic limits. The difference between the analytic limit and the Allowable Value allows for channel instrument accuracy, calibration accuracy, process measurement accuracy, and primary element accuracy. The margin between the Allowable Value and the NTSP allows for instrument drift that might occur during the established surveillance period. Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift. These two verifications are statistical evaluations to provide additional assurance of the acceptability of the NTSP and may require changes to the NTSP. Use of these methods and verifications provides the assurance that if the setpoint is found conservative to the Allowable Value during surveillance testing, the instrumentation would have provided the required trip function by the time the process reached the analytic limit for the applicable events.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Reactor Vessel Water Level - Low Low ~~Low, Level 1~~ signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. 3

The Reactor Vessel Water Level - Low Low ~~Low, Level 1~~ Allowable Value is chosen to allow time for the low pressure core flooding systems to activate and provide adequate cooling. 3

Four channels of CS Reactor Vessel Water Level - Low Low ~~Low, Level 1~~ Function are only required to be OPERABLE when ~~the ECCS is~~ CS and the EDGs are required to be OPERABLE to ensure that no single instrument failure can preclude ~~ECCS~~ these initiation. Per Footnote (a) to Table 3.3.5.1-1, ~~this~~ ECCS Functions ~~is~~ are only required to be OPERABLE in MODES 4 and 5 whenever the associated ECCS is required to be OPERABLE per LCO 3.5.2. Refer to LCO 3.5.1 and LCO 3.5.2~~, "ECCS - Shutdown,"~~ for Applicability Bases for the low pressure ECCS subsystems; LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown," for Applicability Bases for the DGs. 1 1 1 2 1

CS and EDG

Four channels of the LPCI Reactor Vessel Water Level - Low Low Function are only required to be OPERABLE when LPCI is required to be OPERABLE to ensure that no single instrument failure can preclude LPCI initiation.

E 3 8

1.b, 2.b. Drywell Pressure – High

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS and associated E DGs are initiated upon receipt of the Drywell Pressure - High Function in order to minimize the possibility of fuel damage. The Drywell Pressure - High Function, along with the Reactor Water Level - Low Low ~~Low, Level 1~~ Function, is directly assumed in the analysis of the recirculation line break (Ref. ~~4~~ 1). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. 1 3 1

High drywell pressure signals are initiated from four pressure ~~transmitters~~ switches that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment. 1

The Drywell Pressure - High Function is required to be OPERABLE when the ECCS or E DG is required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE. Thus, four channels of the CS and LPCI Drywell Pressure - High Functions are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude ECCS and E DG initiation. In MODES 4 1 5 8 1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

and 5, the Drywell Pressure - High Function s are ~~is~~ not required, since there is insufficient energy in the reactor to pressurize the primary containment to Drywell Pressure - High setpoint. Refer to LCO 3.5.1 for Applicability Bases for the low pressure ECCS subsystems and to LCO 3.8.1 for Applicability Bases for the E DGs. 5 1

1.c, 2.c. Reactor Steam Dome Pressure - Low (Injection Permissive)

Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. This ensures that, prior to opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure. The Reactor Steam Dome Pressure - Low (Injection Permissive) is one of the Functions assumed to be OPERABLE and capable of permitting initiation of the ECCS during the transients analyzed in Reference ~~s 1 and 3~~ 2. In addition, the Reactor Steam Dome Pressure - Low (Injection Permissive) Function is directly assumed in the analysis of the recirculation line break (Ref. ~~2~~ 1). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. 5 1 5 1

The Reactor Steam Dome Pressure - Low (Injection Permissive) signals are initiated from ~~four~~ two (shared by both CS and LPCI) pressure ~~transmitters~~ switches that sense the reactor dome pressure. 5 1

The Allowable Value is low enough to prevent ~~overpressuring~~ overpressurizing the equipment in the low pressure ECCS, but high enough to ensure that the ECCS injection prevents the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46. 4

3 4 ~~Four~~ Two channels of CS Reactor Steam Dome Pressure - Low (Injection Permissive) Function are only required to be OPERABLE when ~~the ECCS~~ CS is required to be OPERABLE to ensure that no single instrument failure can preclude ~~ECCS~~ CS initiation. ~~Per Footnote (a) to Table 3.3.5.1-1, this~~ these ECCS Function ~~is~~ s are only required to be OPERABLE in MODES 4 and 5 whenever the associated ECCS is required to be OPERABLE per LCO 3.5.2. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems. 1

INSERT 7 3

Two channels of the LPCI Reactor Steam Dome Pressure - Low (Injection Premissive) Function are only required to be OPERABLE when LPCI is required to be OPERABLE to ensure that no single instrument failure can preclude LPCI initiation. 1

Insert from Page B 3.3.5.1-12 and B 3.3.5.1.13 3

3 **INSERT 7**

1.d, 2.d. Reactor Steam Dome Pressure - Low (Pump Permissive)

Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. These channels delay CS and LPCI pump starts on Reactor Vessel Water Level - Low Low until reactor steam dome pressure is below the setpoint. This ensures that, prior to starting the pumps of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure. The Reactor Steam Dome Pressure - Low (Pump Permissive) is one of the Functions assumed to be OPERABLE and capable of permitting initiation of the ECCS during the transients analyzed in References 1 and 3. In addition, the Reactor Steam Dome Pressure - Low Function is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The Reactor Steam Dome Pressure - Low signals are initiated from two pressure switches (shared by both CS and LPCI) that sense the reactor dome pressure.

The Allowable Value is high enough to ensure that the ECCS injection prevents the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46.

Two channels of CS Reactor Steam Dome Pressure - Low (Pump Permissive) Function are only required to be OPERABLE when the CS is required to be OPERABLE to ensure that no single instrument failure can preclude CS initiation. Two channels of LPCI Reactor Steam Dome Pressure - Low (Pump Permissive) Function are only required to be OPERABLE when the LPCI is required to be OPERABLE to ensure that no single instrument failure can preclude LPCI initiation. Per Footnote (a) to Table 3.3.5.1-1, these ECCS Functions are only required to be OPERABLE in MODES 4 and 5 whenever the associated ECCS is required to be OPERABLE per LCO 3.5.2. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

1.e, 2.e. Reactor Steam Dome Pressure - Bypass Timer (Pump Permissive)

Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. The Bypass Timer channels allow the CS and LPCI pumps to start on Reactor Vessel Water Level - Low Low after the time delay times out, even if the reactor steam dome pressure is above its permissive setpoint. This ensures that, starting the pumps of the low pressure ECCS subsystems will occur on a Reactor Vessel Water Level - Low Low signal after an approximately 20 minute time delay. The Reactor Steam Dome Pressure - Time Delay (Pump Permissive) is one of the Functions assumed to be OPERABLE and capable of permitting initiation of the ECCS during the transients analyzed in References 1 and 3. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

3 **INSERT 7 (continued)**

The Reactor Steam Dome Pressure - Bypass Timer (Pump Permissive) signals are initiated from four time delay relays.

The Allowable Value is long enough to provide sufficient time for the operator to inhibit any unnecessary ADS actuation, yet short enough to limit the peak cladding temperature to approximately 1592°F.

Two channels of CS Reactor Steam Dome Pressure - Bypass Timer (Pump Permissive) Function are only required to be OPERABLE when the CS is required to be OPERABLE to ensure that no single instrument failure can preclude CS initiation. Two channels of LPCI Reactor Steam Dome Pressure - Bypass Timer (Pump Permissive) Function are only required to be OPERABLE when the LPCI is required to be OPERABLE to ensure that no single instrument failure can preclude LPCI initiation. Per Footnote (a) to Table 3.3.5.1-1, these ECCS Functions are only required to be OPERABLE in MODES 4 and 5 whenever the associated ECCS is required to be OPERABLE per LCO 3.5.2. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~1.d.~~ 2.g. ~~Core Spray and~~ Low Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)

The minimum flow instruments are provided to protect the associated ~~low pressure ECCS~~ pump from overheating when the pump is operating and the associated injection valve is not fully open. The minimum flow line valve is opened when low flow is sensed, and the valve is automatically closed when the flow rate is adequate to protect the pump. The LPCI ~~and CS~~ Pump Discharge Flow - Low Function~~s are~~ assumed to be OPERABLE and capable of closing the minimum flow valves to ensure that the ~~low pressure ECCS~~ flows assumed during the transients and accidents analyzed in References 1, ~~2,~~ and ~~3~~ are met. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

One flow ~~transmitter~~ per ~~ECCS~~ pump is used to detect the associated subsystems' flow rates. The logic is arranged such that each ~~transmitter~~ causes its associated minimum flow valve to open. The logic will close the minimum flow valve once the closure setpoint is exceeded. The LPCI minimum flow valves are time delayed such that the valves will not open for 10 seconds after ~~the switches detect low flow~~. ~~The time delay is provided to limit reactor vessel inventory loss during the startup of the RHR shutdown cooling mode.~~ The Pump Discharge Flow - Low Allowable Values are high enough to ensure that the pump flow rate is sufficient to protect the pump, yet low enough to ensure that the closure of the minimum flow valve is initiated to allow full flow into the core.

Each channel of Pump Discharge Flow - Low Function ~~(two CS channels and~~ four LPCI channels) is only required to be OPERABLE when the associated ~~ECCS~~ is required to be OPERABLE to ensure that no single instrument failure can preclude the ~~ECCS~~ function. Per Footnote (a) to Table 3.3.5.1-1, this ~~ECCS~~ Function is only required to be OPERABLE in MODES 4 and 5 whenever the associated ~~ECCS~~ is required to be OPERABLE per LCO 3.5.2. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the ~~low pressure ECCS~~ subsystems.

~~1.e, 2.h. Manual Initiation~~

~~The Manual Initiation push button channels introduce signals into the appropriate ECCS logic to provide manual initiation capability and are redundant to the automatic protective instrumentation. There is one push button for each of the CS and LPCI subsystems (i.e., two for CS and two for LPCI).~~

1

INSERT 7A

This delay can reduce the reactor vessel inventory loss (to the suppression pool) during the startup of the RHR pump while aligned in the shutdown cooling mode, since it provides time (prior to opening the minimum flow valve) to manually increase RHR flow above the minimum flow closure setpoint.

3

INSERT 8

2.h, 2.k. Reactor Steam Dome Pressure - Low (Break Detection) and Reactor Steam Dome Pressure - Time Delay Relay (Break Detection)

The purpose of the Reactor Steam Dome Pressure - Low (Break Detection) and Reactor Steam Dome Pressure - Time Delay Relay (Break Detection) Functions are to optimize the LPCI Loop Select Logic sensitivity if the logic previously actuated recirculation pump trips. This is accomplished by preventing the logic from continuing on to the unbroken loop selection activity until reactor steam dome pressure has dropped below a specified value. These Functions are only required to be OPERABLE for the DBA LOCA analysis, i.e., if the break location is in the recirculation system suction piping (Ref. 2). For a DBA LOCA, the analysis assumes that the LPCI Loop Select Logic successfully identifies and directs LPCI flow to the unbroken recirculation loop so that core reflooding is accomplished in time to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. For other LOCA events, (i.e., non-DBA recirculation system pipe breaks), or other RPV pipe breaks the success of the Loop Select Logic is less critical than for the DBA.

Reactor Steam Dome Pressure - Low (Break Detection) signals are initiated from four pressure switches that sense the reactor steam dome pressure. Reactor Steam Dome Pressure - Time Delay Relay (Break Detection) signals are initiated from two time delay relays.

The Reactor Steam Dome Pressure - Low (Break Detection) Allowable Value is chosen to allow for coastdown of any recirculation pump which has just tripped, thus optimizing the sensitivity of the LPCI Loop Select Logic while ensuring that LPCI injection is not delayed. The Reactor Steam Dome Pressure - Time Delay Relay (Break Detection) Allowable Value is chosen to allow momentum effects to establish the maximum differential pressure for break detection.

Four channels of the Reactor Steam Dome Pressure - Low (Break Detection) Function and two channels of the Reactor Steam Dome Pressure - Time Delay Relay (Break Detection) Function are only required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single failure can prevent the LPCI Loop Select Logic from successfully selecting the unbroken recirculation loop for LPCI injection. These Functions are not required to be OPERABLE in MODES 4 and 5 because, in those MODES, the loop for selection is controlled by plant operating procedures, which ensure an OPERABLE LPCI flow path.

1 **INSERT 8 (continued)**

2.i, 2.l. Recirculation Pump Differential Pressure - High (Break Detection) and Recirculation Pump Differential Pressure - Time Delay Relay (Break Detection)

Recirculation pump differential pressure signals are used by the LPCI Loop Select Logic to determine if either recirculation pump is running. If either pump is not running, i.e., single loop operation, the logic, after a short time delay, sends a trip signal to both recirculation pumps. This is necessary to eliminate the possibility of small pipe breaks being masked by a running recirculation pump. These Functions are only required to be OPERABLE for the DBA LOCA analysis, i.e., if the break location is in the recirculation system suction piping (Ref. 2). For a DBA LOCA, the analysis assumes that the LPCI Loop Select Logic successfully identifies and directs LPCI flow to the unbroken recirculation loop so that core reflooding is accomplished in time to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. For other LOCA events (i.e., non-DBA recirculation system pipe breaks or other RPV pipe breaks), the success of the Loop Select Logic is less critical than for the DBA.

Recirculation Pump Differential Pressure - High (Break Detection) signals are initiated from eight differential pressure switches, four of which sense the pressure differential between the suction and discharge of each recirculation pump. Recirculation Pump Differential Pressure - Time Delay Relay (Break Detection) signals are initiated by two time delay relays.

The Recirculation Pump Differential Pressure - High (Break Detection) Allowable Value is chosen to be as low as possible, while still maintaining the ability to differentiate between a running and non-running recirculation pump. Recirculation Pump Differential Pressure - Time Delay Relay (Break Detection) Allowable Value is chosen to allow enough time to determine the status of the operating conditions of the recirculation pumps.

Eight channels of the Recirculation Pump Differential Pressure - High (Break Detection) Function and two channels of the Recirculation Pump Differential Pressure - Time Delay Relay (Break Detection) Function are only required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single failure can prevent the LPCI Loop Select Logic from successfully determining if either recirculation pump is running. This Function is not required to be OPERABLE in MODES 4 and 5 because, in those MODES, the loop for selection is controlled by plant operating procedures, which ensure an OPERABLE LPCI flow path.

1

INSERT 8 (continued)

2.j, 2.m. Recirculation Riser Differential Pressure - High (Break Detection) and Recirculation Riser Differential Pressure - Time Delay Relay (Break Detection)

Recirculation riser differential pressure signals are used by the LPCI Loop Select Logic to determine which, if any, recirculation loop is broken. This is accomplished by comparing the pressure of the two recirculation loops. A broken loop will be indicated by a lower pressure than an unbroken loop. The loop with the higher pressure is then selected, after a short delay, for LPCI injection. If neither loop is broken, the logic defaults to injecting into the "B" recirculation loop. These Functions are only required to be OPERABLE for the DBA LOCA analysis, i.e., if the break location is in the recirculation system suction piping (Ref. 2). For a DBA LOCA, the analysis assumes that the LPCI Loop Select Logic successfully identifies and directs LPCI flow to the unbroken recirculation loop, so that core reflooding is accomplished in time to ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. For other LOCA events, (i.e., non-DBA recirculation system pipe breaks), or other RPV pipe breaks, the success of the Loop Select Logic is less critical than for the DBA.

Recirculation Riser Differential Pressure - High (Break Detection) signals are initiated from four differential pressure switches that sense the pressure differential between the A recirculation loop riser and the B recirculation loop riser. If, after a small time delay, the pressure in loop A is not indicating higher than loop B pressure, the logic will select the B loop for injection. If recirculation loop A pressure is indicating higher than loop B pressure, the logic will select the A loop for LPCI injection. Recirculation Riser Differential Pressure - Time Delay - Relay (Break Detection) signals are initiated by two time delay relays.

The Recirculation Riser Differential Pressure - High (Break Detection) Allowable Value is chosen to be as low as possible, while still maintaining the ability to differentiate between a broken and unbroken recirculation loop. The Recirculation Riser Differential Pressure - Time Delay Relay (Break Detection) Allowable Value is chosen to provide a sufficient amount of time to determine which loop is broken.

Four channels of the Recirculation Riser Differential Pressure - High (Break Detection) Function and two channels of the Recirculation Riser Differential Pressure - Time Delay Relay (Break Detection) Function are only required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single failure can prevent the LPCI Loop Select Logic from successfully selecting the unbroken recirculation loop for LPCI injection. This Function is not required to be OPERABLE in MODES 4 and 5 because, in those MODES, the loop for selection is controlled by plant operating procedures, which ensure an OPERABLE LPCI flow path.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Manual Initiation Function is not assumed in any accident or transient analyses in the FSAR. However, the Function is retained for overall redundancy and diversity of the low pressure ECCS function as required by the NRC in the plant licensing basis.

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons. Each channel of the Manual Initiation Function (one channel per subsystem) is only required to be OPERABLE when the associated ECCS is required to be OPERABLE. Per Footnote (a) to Table 3.3.5.1-1, this ECCS Function is only required to be OPERABLE in MODES 4 and 5 whenever the associated ECCS is required to be OPERABLE per LCO 3.5.2. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems. 3

<u>2.d. Reactor Steam Dome Pressure - Low (Recirculation Discharge Valve Permissive)</u>

Low reactor steam dome pressure signals are used as permissives for recirculation discharge valve closure. This ensures that the LPCI subsystems inject into the proper RPV location assumed in the safety analysis. The Reactor Steam Dome Pressure - Low is one of the Functions assumed to be OPERABLE and capable of closing the valve during the transients analyzed in References 1 and 3. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Steam Dome Pressure - Low Function is directly assumed in the analysis of the recirculation line break (Ref. 2). 3

The Reactor Steam Dome Pressure - Low signals are initiated from four pressure transmitters that sense the reactor dome pressure.

The Allowable Value is chosen to ensure that the valves close prior to commencement of LPCI injection flow into the core, as assumed in the safety analysis.

Four channels of the Reactor Steam Dome Pressure - Low Function are only required to be OPERABLE in MODES 1, 2, and 3 with the associated recirculation pump discharge valve open. With the valve(s) closed, the function instrumentation has been performed; thus, the Function is not required. In MODES 4 and 5, the loop injection location is not critical since LPCI injection through the recirculation loop in either direction will still ensure that LPCI flow reaches the core (i.e., there is no significant reactor steam dome back pressure).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.e. Reactor Vessel Shroud Level - Level 0

The Level 0 Function is provided as a permissive to allow the RHR System to be manually aligned from the LPCI mode to the suppression pool cooling/spray or drywell spray modes. The permissive ensures that water in the vessel is approximately two thirds core height before the manual transfer is allowed. This ensures that LPCI is available to prevent or minimize fuel damage. This function may be overridden during accident conditions as allowed by plant procedures. Reactor Vessel Shroud Level - Level 0 Function is implicitly assumed in the analysis of the recirculation line break (Ref. 2) since the analysis assumes that no LPCI flow diversion occurs when reactor water level is below Level 0.

Reactor Vessel Shroud Level - Level 0 signals are initiated from two level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. The Reactor Vessel Shroud Level - Level 0 Allowable Value is chosen to allow the low pressure core flooding systems to activate and provide adequate cooling before allowing a manual transfer. 3

Two channels of the Reactor Vessel Shroud Level - Level 0 Function are only required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the specified initiation time of the LPCI subsystems is not assumed, and other administrative controls are adequate to control the valves that this Function isolates (since the systems that the valves are opened for are not required to be OPERABLE in MODES 4 and 5 and are normally not used).

move to page B 3.3.5.1-9 as indicated 3

Core Spray and 1.f, 2.f. Low Pressure Coolant Injection Pump Start - Time Delay Relay 3

e relays CS and

The purpose of this time delay is to stagger the start of the LPCI pumps that are in each of Divisions 1 and 2, thus limiting the starting transients on the 4.16 kV ~~emergency~~ buses. This Function is only necessary when power is being supplied from the standby power sources (DG). However, since the time delay does not degrade ECCS operation, it remains in the pump start logic at all times. The LPCI Pump Start - Time Delay Relays are assumed to be OPERABLE in the accident and transient analyses requiring ECCS initiation. That is, the analyses assume that the pumps will initiate when required and excess loading will not cause failure of the power sources. 3 1 3

essential CS and

move to page B 3.3.5.1-9 as indicated 3

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

two CS Pump Start - Time Delay Relay channels 3 CS CS Pump Start - Time Delay Relay channel from 4 two are 3 s CS and

There are ~~four LPCI Pump Start - Time Delay Relays~~, one in each of the CS pump 3 ~~RHR~~ pump start logic circuits. While each time delay relay is dedicated to a single pump start logic, a single failure of a ~~LPCI Pump Start - Time Delay Relay~~ three 1 3 could result in the failure of the ~~two~~ low pressure ECCS 4.16 kV essential pumps, powered ~~for~~ the same ~~ESF~~ bus, to perform their intended function ~~within the assumed ECCS RESPONSE TIME~~ (e.g., as in the case where 3 ~~both~~ ECCS pumps on one ~~ESF~~ bus start simultaneously due to an 4.16 kV essential 1 inoperable time delay relay). This still leaves ~~four~~ of the six low pressure three ECCS pumps OPERABLE; thus, the single failure criterion is met (i.e., INSERT 8A loss of one instrument does not preclude ECCS initiation). ~~The Allowable Value for the LPCI Pump Start - Time Delay Relays is chosen to be long enough so that most of the starting transient of the first pump is complete before starting the second pump on the same 4.16 kV emergency bus and short enough so that ECCS operation is not degraded.~~ 3

CS and Each LPCI Pump Start - Time Delay Relay Function is required to be ECCS OPERABLE only when the associated ~~LPCI~~ subsystem is required to be these OPERABLE. Per Footnote (a) to Table 3.3.5.1-1, ~~this~~ ECCS Function ~~is~~ s are only required to be OPERABLE in MODES 4 and 5 whenever the associated ECCS is required to be OPERABLE per LCO 3.5.2. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the LPCI subsystems. 3

HPCI System

3.a. Reactor Vessel Water Level - Low Low~~, Level 2~~ 3

Low Low

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the HPCI System is initiated at ~~Level 2~~ to maintain level above the top of the active fuel. The Reactor Vessel Water Level - Low Low~~, Level 2~~ is one of the Functions assumed to be OPERABLE and capable of initiating HPCI during the transients analyzed in Reference~~s 1 and 3~~ 2. Additionally, the Reactor Vessel Water Level - Low Low~~, Level 2~~ Function associated with HPCI along with the Drywell Pressure - High Function is directly assumed in the analysis of the recirculation line break (Ref. ~~2~~ 1). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. 3 1 3 1

Reactor Vessel Water Level - Low Low~~, Level 2~~ signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. 3

INSERT 8A

Sixteen Low Pressure Coolant Injection Pump Start - Time Delay Relay channels, four in each of the LPCI pump start logic circuits, are required to be OPERABLE to ensure that no single instrument failure can preclude the associated LPCI pump start.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

when the reactor vessel is isolated

The Reactor Vessel Water Level - Low Low, ~~Level 2~~ Allowable Value is high enough such that for complete loss of feedwater flow, the Reactor Core Isolation Cooling (RCIC) System flow with HPCI assumed to fail will be sufficient to avoid ~~initiation~~ of low pressure ECCS ~~at Reactor Vessel Water Level - Low Low Low, Level 1~~. (3) (1) (1)

Injection

Four channels of Reactor Vessel Water Level - Low Low, ~~Level 2~~ Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases. (3)

3.b. Drywell Pressure – High

High pressure in the drywell could indicate a break in the RCPB. The HPCI System is initiated upon receipt of the Drywell Pressure - High Function in order to minimize the possibility of fuel damage. The Drywell Pressure - High Function, along with the Reactor Water Level - Low Low, ~~Level 2~~ Function, is directly assumed in the analysis of the recirculation line break (Ref. ~~4~~). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. (3)

(5) Vessel

1

switches

High drywell pressure signals are initiated from four pressure ~~transmitters~~ that sense drywell pressure. The Allowable Value was selected to be as low as possible to be indicative of a LOCA inside primary containment. (3)

Four channels of the Drywell Pressure - High Function are required to be OPERABLE when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for the Applicability Bases for the HPCI System.

3.c. Reactor Vessel Water Level - High~~, Level 8~~ (3)

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the ~~Level 8~~ signal is used to trip the HPCI turbine to prevent overflow into the main steam lines (MSLs). The Reactor Vessel Water Level - High~~, Level 8~~ Function is not assumed in the accident and transient analyses. It was retained since it is a potentially significant contributor to risk. (3) (3)

Reactor Vessel Water Level - High

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Reactor Vessel Water Level - High, ~~Level 8~~ signals for HPCI are initiated from two level transmitters from the narrow range water level measurement instrumentation. Both ~~Level 8~~ signals are required in order to close the HPCI ~~injection~~ valve. This ensures that no single instrument failure can preclude HPCI initiation. The Reactor Vessel Water Level - High, ~~Level 8~~ Allowable Value is chosen to prevent flow from the HPCI System from overflowing into the MSLs. 3 3 turbine's stop 1 3

Two channels of Reactor Vessel Water Level - High, ~~Level 8~~ Function are required to be OPERABLE only when HPCI is required to be OPERABLE. Refer to LCO 3.5.1 ~~and LCO 3.5.2~~ for HPCI Applicability Bases. 3 4

3.d. Condensate Storage Tank Level – Low

Low level in ~~the~~ CST indicates the unavailability of an adequate supply of makeup water from this normal source. Normally the suction valves between HPCI and the CST are open and, upon receiving a HPCI initiation signal, water for HPCI injection would be taken from ~~the~~ CST. However, if the water level in ~~the~~ CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the HPCI pump. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes. The Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool. a all aligned s s any 1 s 1

Condensate Storage Tank Level - Low signals are initiated from two level switches. The logic is arranged such that either level switch can cause the suppression pool suction valves to open and the CST suction valve to close. The Condensate Storage Tank Level - Low Function Allowable Value is high enough to ensure adequate pump suction head while water is being taken from the CST. The Allowable Value is referenced from the bottom of the tank. (normally one associated with each CST) s s 1 1 1

Two channels of the Condensate Storage Tank Level - Low Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI swap to suppression pool source. Refer to LCO 3.5.1 for HPCI Applicability Bases.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3.e. Suppression Pool Water Level – High

Excessively high suppression pool water could result in the loads on the suppression pool exceeding design values should there be a blowdown of the reactor vessel pressure through the safety/relief valves. Therefore, signals indicating high suppression pool water level are used to transfer the suction source of HPCI from the CST to the suppression pool to eliminate the possibility of HPCI continuing to provide additional water from a source outside containment. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes. This Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool.

Suppression Pool Water Level - High signals are initiated from two level switches. The logic is arranged such that either switch can cause the suppression pool suction valves to open and the CST suction valve to close. The Allowable Value for the Suppression Pool Water Level - High Function is chosen to ensure that HPCI will be aligned for suction from the suppression pool before the water level reaches the point at which suppression pool design loads would be exceeded.

Two channels of Suppression Pool Water Level - High Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI swap to suppression pool source. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.f. High Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)

The minimum flow instruments are provided to protect the HPCI pump from overheating when the pump is operating and the associated injection valve is not fully open. The minimum flow line valve is opened when low flow is sensed, and the valve is automatically closed when the flow rate is adequate to protect the pump. The High Pressure Coolant Injection Pump Discharge Flow - Low (Bypass) Function is assumed to be OPERABLE and capable of closing the minimum flow valve to ensure that the ECCS flow assumed during the transients and accidents analyzed in References 1, 2, and 3 are met. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

One flow ~~transmitter~~ switch is used to detect the HPCI System's flow rate. The logic is arranged such that the ~~transmitter~~ switch causes the minimum flow valve to open. The logic will close the minimum flow valve once the closure setpoint is exceeded. 1

The High Pressure Coolant Injection Pump Discharge Flow - Low (Bypass) Allowable Value is high enough to ensure that pump flow rate is sufficient to protect the pump, yet low enough to ensure that the closure of the minimum flow valve is initiated to allow full flow into the core. 5

One channel is required to be OPERABLE when the HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

~~3.g. Manual Initiation~~

~~The Manual Initiation push button channel introduces signals into the HPCI logic to provide manual initiation capability and is redundant to the automatic protective instrumentation. There is one push button for the HPCI System.~~

~~The Manual Initiation Function is not assumed in any accident or transient analyses in the FSAR. However, the Function is retained for overall redundancy and diversity of the HPCI function as required by the NRC in the plant licensing basis.~~ 3

~~There is no Allowable Value for this Function since the channel is mechanically actuated based solely on the position of the push button. One channel of the Manual Initiation Function is required to be OPERABLE only when the HPCI System is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.~~

Automatic Depressurization System

4.a, 5.a. Reactor Vessel Water Level - Low Low ~~Low, Level 1~~ 3

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, ADS receives one of the signals necessary for initiation from this Function. The Reactor Vessel Water Level - Low Low ~~Low, Level 1~~ is one of the Functions assumed to be OPERABLE and capable of initiating the ADS during the accident analyzed in Reference ~~2~~ 1. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. 3 1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Reactor Vessel Water Level - Low Low Low Level 1 signals are initiated 3
from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Low Low Level 1 Function 3
are required to be OPERABLE only when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

The Reactor Vessel Water Level - Low Low Low Level 1 Allowable Value 3
is chosen to allow time for the low pressure core flooding systems to initiate and provide adequate cooling.

4.b, 5.b. Drywell Pressure – High

High pressure in the drywell could indicate a break in the RCPB. Therefore, ADS receives one of the signals necessary for initiation from this Function in order to minimize the possibility of fuel damage. The Drywell Pressure - High is assumed to be OPERABLE and capable of initiating the ADS during the accidents analyzed in Reference 2. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the 3
limits of 10 CFR 50.46.

Drywell Pressure - High signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

Four channels of Drywell Pressure - High Function are only required to be OPERABLE when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4.c, 5.c. Automatic Depressurization System Initiation Timer (3)

The purpose of the Automatic Depressurization System Initiation Timer is to delay depressurization of the reactor vessel to allow the HPCI System time to maintain reactor vessel water level. Since the rapid depressurization caused by ADS operation is one of the most severe transients on the reactor vessel, its occurrence should be limited. By delaying initiation of the ADS Function, the operator is given the chance to monitor the success or failure of the HPCI System to maintain water level, and then to decide whether ~~or not~~ to allow ADS to initiate, to delay initiation ~~further by recycling the timer, or to inhibit initiation permanently~~. (1) The Automatic Depressurization System Initiation Timer Function is assumed to be OPERABLE for the accident analyses of Reference 2 that require ECCS initiation and assume failure of the HPCI System. (1)

automatically

or

or inhibit ADS

b

1

There are two Automatic Depressurization System Initiation Timer relays, one in each of the two ADS trip systems. The Allowable Value for the Automatic Depressurization System Initiation Timer is chosen so that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

Two channels of the Automatic Depressurization System Initiation Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases. (4)

~~4.d, 5.d. Reactor Vessel Water Level - Low, Level 3~~

~~The Reactor Vessel Water Level - Low, Level 3 Function is used by the ADS only as a confirmatory low water level signal. ADS receives one of the signals necessary for initiation from Reactor Vessel Water Level - Low Low Low, Level 1 signals. In order to prevent spurious initiation of the ADS due to spurious Level 1 signals, a Level 3 signal must also be received before ADS initiation commences.~~ (3)

~~Reactor Vessel Water Level - Low, Level 3 signals are initiated from two level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. The Allowable Value for Reactor Vessel Water Level - Low, Level 3 is selected at the RPS Level 3 scram Allowable Value for convenience. Refer to LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," for the Bases discussion of this Function.~~

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~Two channels of Reactor Vessel Water Level - Low, Level 3 Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.~~ 3

c d c d

4.e, 4.f, 5.e, 5.f. Core Spray and Low Pressure Coolant Injection Pump Discharge Pressure – High 3

The Pump Discharge Pressure - High signals from the CS and LPCI pumps are used as permissives for ADS initiation, indicating that there is a source of low pressure cooling water available once the ADS has depressurized the vessel. Pump Discharge Pressure - High is one of the Functions assumed to be OPERABLE and capable of permitting ADS 1 initiation during the events analyzed in Reference 2 with an assumed HPCI failure. For these events the ADS depressurizes the reactor vessel so that the low pressure ECCS can perform the core cooling functions. This core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. 1

Pump discharge pressure signals are initiated from twelve pressure switches ~~transmitters~~, two on the discharge side of each of the six low pressure ECCS pumps. In order to generate an ADS permissive in one trip system, it is necessary that only one pump (both channels for the pump) indicate the high discharge pressure condition. The Pump Discharge Pressure - High Allowable Value is less than the pump discharge pressure when the pump is operating in a full flow mode and high enough to avoid any condition that results in a discharge pressure permissive when the CS and LPCI pumps are aligned for injection and the pumps are not running. The actual operating point of this function is not assumed in any transient or accident analysis. 1

Twelve channels of Core Spray and Low Pressure Coolant Injection Pump Discharge Pressure - High Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two CS channels associated with CS pump A and four LPCI channels associated with LPCI pumps A and C ~~D~~ are required for trip system A. Two CS channels associated with CS pump B and four LPCI channels associated with LPCI pumps B and D ~~C~~ are required for trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases. 1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4.g, 5.g. Automatic Depressurization System Low Water Level Actuation Timer

One of the signals required for ADS initiation is Drywell Pressure - High. However, if the event requiring ADS initiation occurs outside the drywell (e.g., main steam line break outside containment), a high drywell pressure signal may never be present. Therefore, the Automatic Depressurization System Low Water Level Actuation Timer is used to bypass the Drywell Pressure - High Function after a certain time period has elapsed. Operation of the Automatic Depressurization System Low Water Level Actuation Timer Function is not assumed in any accident analysis. The instrumentation is retained in the TS because ADS is part of the primary success path for mitigation of a DBA. 3

There are four Automatic Depressurization System Low Water Level Actuation Timer relays, two in each of the two ADS trip systems. The Allowable Value for the Automatic Depressurization System Low Water Level Actuation Timer is chosen to ensure that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

Four channels of the Automatic Depressurization System Low Water Level Actuation Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.h, 5.h. Manual Initiation

The Manual Initiation push button channels introduce signals into the ADS logic to provide manual initiation capability and are redundant to the automatic protective instrumentation. There are two push buttons for each ADS trip system for a total of four.

The Manual Initiation Function is not assumed in any accident or transient analyses in the FSAR. However, the Function is retained for overall redundancy and diversity of the ADS functions as required by the NRC in the plant licensing basis. 3

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons. Four channels of the Manual Initiation Function (two channels per trip system) are only required to be OPERABLE when the ADS is required to be OPERABLE. Refer to LCO 3.5.1 for ADS Applicability Bases.

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~~-----REVIEWER'S NOTE-----~~
~~Certain LCO Completion Times are based on approved topical reports. In order for a licensee to use the times, the licensee must justify the Completion Times as required by the staff Safety Evaluation Report (SER) for the topical report.~~ 6

A Note has been provided to modify the ACTIONS related to ECCS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable ECCS instrumentation channels provide appropriate compensatory measures for separate inoperable Condition entry for each inoperable ECCS instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.5.1-1. The applicable Condition referenced in the Table is Function dependent. Each time a channel is discovered inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

B.1, B.2, and B.3

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in redundant automatic initiation capability being lost for the feature(s). Required Action B.1 features would be those that are i.e. initiated by Functions 1.a, 1.b, 2.a, ~~and~~ 2.b, ~~(e.g.~~, low pressure ECCS~~)~~. 3 The Required Action B.2 system would be HPCI. For Required 2.f, 2.h, and 2.k Action B.1, redundant automatic initiation capability is lost if ~~(a) two~~ and associated EDG 1 ~~Function 1.a channels are inoperable and untripped in the same trip system, (b) two Function 2.a channels are inoperable and untripped in the same trip system, (c) two Function 1.b channels are inoperable and untripped in the same system, or (d) two Function 2.b channels are inoperable and untripped in the same trip system.~~ INSERT 9 1 For low pressure

1 **INSERT 9**

: (a) two or more Function 1.a channels are inoperable and untripped such that both trip systems lose initiation capability; (b) two or more Function 2.a channels are inoperable and untripped such that both trip systems lose initiation capability; (c) two or more Function 1.b channels are inoperable and untripped such that both trip systems lose initiation capability; (d) two or more Function 2.b channels are inoperable and untripped such that both trip systems lose initiation capability; (e) two or more Function 2.f channels are inoperable and untripped such that one or more pumps in both LPCI subsystems lose initiation (i.e., time delay) capability; (f) two or more Function 2.h channels are inoperable and untripped such that both trip systems lose initiation capability; or (g) two Function 2.k channels are inoperable and untripped.

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ACTIONS (continued)

ECCS, since each inoperable channel would have Required Action B.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated system of low pressure ECCS and DGs to be declared inoperable. However, since channels in both associated low pressure ECCS subsystems (e.g., both CS subsystems) are inoperable and untripped, and the Completion Times started concurrently for the channels in both subsystems, this results in the affected portions in the associated low pressure ECCS and DGs being concurrently declared inoperable. E 1 1

(a trip system in this case is defined as channels associated with the parallel level in the logic arrangement)

For Required Action B.2, redundant automatic initiation capability is lost if two Function 3.a or two Function 3.b channels are inoperable and untripped in the same trip system. In this situation (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action B.3 is not appropriate and the feature(s) associated with the inoperable, untripped channels must be declared inoperable within 1 hour. As noted (Note 1 to Required Action B.1), Required Action B.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the low pressure ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of initiation capability for 24 hours (as allowed by Required Action B.3) is allowed during MODES 4 and 5. There is no similar Note provided for Required Action B.2 since HPCI instrumentation is not required in MODES 4 and 5; thus, a Note is not necessary. 1 2 2

Notes are also provided (Note 2 to Required Action B.1 and the Note to Required Action B.2) to delineate which Required Action is applicable for each Function that requires entry into Condition B if an associated channel is inoperable. This ensures that the proper loss of initiation capability check is performed. ~~Required Action B.1 (the Required Action for certain inoperable channels in the low pressure ECCS subsystems) is not applicable to Function 2.e, since this Function provides backup to administrative controls ensuring that operators do not divert LPCI flow from injecting into the core when needed. Thus, a total loss of Function 2.e capability for 24 hours is allowed, since the LPCI subsystems remain capable of performing their intended function.~~ 3

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action B.1, the Completion Time only begins upon discovery that a redundant feature in the same

BASES

ACTIONS (continued)

system (e.g., both CS subsystems) cannot be automatically initiated due to inoperable, untripped channels within the same Function as described in the paragraph above. For Required Action B.2, the Completion Time only begins upon discovery that the HPCI System cannot be automatically initiated due to two inoperable, untripped channels for the associated Function in the same trip system. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. ~~5~~ 3) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.3. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition H must be entered and its Required Action taken. 1

INSERT 9A 1

C.1 and C.2

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the same Function result in redundant automatic initiation capability being lost for the feature(s). Required Action C.1 features would be those that are initiated by Functions ~~1.c~~ 1.d, 1.e, 1.f, 2.c, ~~2.d, and 2.f~~ 2.e, 2.i, 2.j, 2.l, and 2.m (i.e., low pressure ECCS). Redundant automatic initiation capability is lost if ~~either (a) two Function 1.c channels are inoperable in the same trip system, (b) two Function 2.c channels are inoperable in the same trip system, (c) two Function 2.d channels are inoperable in the same trip system, or (d) two or more Function 2.f channels are inoperable~~ INSERT 10. ¶ In this situation (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action C.2 is not appropriate and the feature(s) associated with the inoperable channels must be declared inoperable within 1 hour. Since each inoperable channel would have Required Action C.1 applied separately (refer to 3 1 2 2

1 **INSERT 9A**

or as in the case where placing an inoperable channel in trip would result in an immediate initiation without time delay when an initiation signal is received

1 **INSERT 10**

: (a) two Function 1.c channels are inoperable; (b) two Function 2.c channels are inoperable; (c) two Function 1.d channels are inoperable; (d) two Function 2.d channels are inoperable; (e) two Function 1.e channels are inoperable; (f) two Function 2.e channels are inoperable; (g) two Function 1.f channels are inoperable; (h) two or more Function 2.i channels, associated with a recirculation pump are inoperable such that both trip systems lose initiation capability; (i) two or more Function 2.j channels are inoperable such that both trip systems lose initiation capability; (j) two Function 2.l channels are inoperable; or (k) two Function 2.m channels are inoperable.

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ACTIONS (continued)

Insert text to page B 3.3.5.1-24 as indicated

ACTIONS Note), each inoperable channel would only require the affected portion of the associated system to be declared inoperable. However, since channels for both low pressure ECCS subsystems are inoperable (e.g., both CS subsystems), and the Completion Times started concurrently for the channels in both subsystems, this results in the affected portions in both subsystems being concurrently declared inoperable. For these Functions ~~1.c, 2.d, and 2.f~~ the affected portions are the associated low pressure ECCS pumps. As noted (Note 1), Required Action C.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of automatic initiation capability for 24 hours (as allowed by Required Action C.2) is allowed during MODES 4 and 5. (2) (1)

Note 2 states that Required Action C.1 is only applicable for Functions 1.c, 1.d, 1.e, 1.f, 2.c, 2.d, 2.e, 2.i, 2.j, 2.l, and 2.m ~~and 2.f~~. Required Action C.1 is not applicable to Functions 1.e, 2.h, and 3.g (which also require entry into this Condition if a channel in these Functions is inoperable), since they are the Manual Initiation Functions and are not assumed in any accident or transient analysis. Thus, a total loss of manual initiation capability for 24 hours (as allowed by Required Action C.2) is allowed. Required Action C.1 is ~~also~~ not applicable to Function 3.c (which also requires entry into this Condition if a channel in this Function is inoperable), since the loss of one channel results in a loss of the Function (two-out-of-two logic). This loss was considered during the development of Reference ~~5~~ 3 and considered acceptable for the 24 hours allowed by Required Action C.2. (3) (3) (3) (1)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action C.1, the Completion Time only begins upon discovery that the same feature in both subsystems (e.g., both CS subsystems) cannot be automatically initiated due to inoperable channels within the same Function as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

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ACTIONS (continued)

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 5) to permit 3 restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would either cause the initiation or it would not necessarily result in a safe state for the channel in all events. 1

D.1, D.2.1, and D.2.2

Required Action D.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic component initiation capability for the HPCI System. Automatic component initiation capability is lost if two Function 3.d channels or two Function 3.e channels are inoperable and untripped. In this situation (loss of automatic suction swap), the 24 hour allowance of Required Actions D.2.1 and D.2.2 is not appropriate and the HPCI System must be declared inoperable within 1 hour after discovery of loss of HPCI initiation capability. As noted, Required Action D.1 is only applicable if the HPCI pump suction is not aligned to the suppression pool, since, if aligned, the Function is already performed.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action D.1, the Completion Time only begins upon discovery that the HPCI System cannot be automatically aligned to the suppression pool due to two inoperable, untripped channels in the same Function. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 5) to permit 3 restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action D.2.1 or the suction source must be aligned to the suppression pool per Required Action D.2.2. Placing the 1

BASES

ACTIONS (continued)

inoperable channel in trip performs the intended function of the channel (shifting the suction source to the suppression pool). Performance of either of these two Required Actions will allow operation to continue. If Required Action D.2.1 or D.2.2 is performed, measures should be taken to ensure that the HPCI System piping remains filled with water. Alternately, if it is not desired to perform Required Actions D.2.1 and D.2.2 (e.g., as in the case where shifting the suction source could drain down the HPCI suction piping), Condition H must be entered and its Required Action taken.

E.1 and E.2

Required Action E.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the ~~Core Spray and~~ Low Pressure Coolant Injection Pump Discharge Flow - Low Bypass Function~~s~~ result in redundant automatic initiation capability being lost for the feature(s). For Required Action E.1, the features would be those that are initiated by Function~~s 1.d and~~ 2.g ~~(e.g., low pressure ECCS)~~. Redundant automatic initiation capability is lost if ~~(a) two Function 1.d channels are inoperable or (b)~~ one or more Function 2.g channels associated with pumps in LPCI subsystem A and one or more Function 2.g channels associated with pumps in LPCI subsystem B are inoperable. Since each inoperable channel would have Required Action E.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected ~~low pressure ECCS~~ pump to be declared inoperable. However, since channels for more than one ~~low pressure ECCS~~ pump are inoperable, and the Completion Times started concurrently for the channels of the ~~low pressure ECCS~~ pumps, this results in the affected low pressure ECCS pumps being concurrently declared inoperable.

(s i.e., LPCI) LPCI

3 5 3 3

In this situation (loss of redundant automatic initiation capability), the 7 day allowance of Required Action E.2 is not appropriate and the subsystem associated with each inoperable channel must be declared inoperable within 1 hour. As noted (Note 1 to Required Action E.1), Required Action E.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of initiation capability for 7 days (as allowed by Required Action E.2) is allowed

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ACTIONS (continued)

during MODES 4 and 5. A Note is also provided (Note 2 to Required Action E.1) to delineate that Required Action E.1 is only applicable to ~~low pressure ECCS~~ the LPCI Function~~s~~. Required Action E.1 is not applicable to HPCI Function 3.f since the loss of one channel results in a loss of the Function (one-out-of-one logic). This loss was considered during the development of Reference ~~5~~ 3 and considered acceptable for the 7 days allowed by Required Action E.2. (3) (1)

~~The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock."~~ (2)

For Required Action E.1, the Completion Time only begins upon discovery that a redundant feature in the same system (~~e.g.~~ i.e., both ~~CS~~ LPCI subsystems) cannot be automatically initiated due to inoperable channels within the same Function as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels. (3)

If the instrumentation that controls the pump minimum flow valve is inoperable, such that the valve will not automatically open, extended pump operation with no injection path available could lead to pump overheating and failure. If there were a failure of the instrumentation, such that the valve would not automatically close, a portion of the pump flow could be diverted from the reactor vessel injection path, causing insufficient core cooling. These consequences can be averted by the operator's manual control of the valve, which would be adequate to maintain ECCS pump protection and required flow. Furthermore, other ECCS pumps would be sufficient to complete the assumed safety function if no additional single failure were to occur. The 7 day Completion Time of Required Action E.2 to restore the inoperable channel to OPERABLE status is reasonable based on the remaining capability of the associated ECCS subsystems, the redundancy available in the ECCS design, and the low probability of a DBA occurring during the allowed out of service time. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

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ACTIONS (continued)

F.1 and F.2

Required Action F.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within similar ADS trip system A and B Functions result in redundant automatic initiation capability being lost for the ADS. Redundant automatic initiation capability is lost if ~~either (a)~~ one Function 4.a channel and one Function 5.a channel are inoperable and untripped ~~, (b) one Function 4.b channel and one Function 5.b channel are inoperable and untripped, or (c) one Function 4.d channel and one Function 5.d channel are inoperable and untripped~~. 3 3

In this situation (loss of automatic initiation capability), the 96 hour or 8 day allowance, as applicable, of Required Action F.2 is not appropriate and all ADS valves must be declared inoperable within 1 hour after discovery of loss of ADS initiation capability.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action F.1, the Completion Time only begins upon discovery that the ADS cannot be automatically initiated due to inoperable, untripped channels within similar ADS trip system Functions as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 8 days has been shown to be acceptable (Ref. ~~5~~ 3) to permit restoration of any inoperable channel to OPERABLE status if both HPCI and RCIC are OPERABLE. If either HPCI or RCIC is inoperable, the time is shortened to 96 hours. If the status of HPCI or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCI or RCIC inoperability. However, the total time for an inoperable, untripped channel cannot exceed 8 days. If the status of HPCI or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable, untripped channel. If the 1

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ACTIONS (continued)

inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action F.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition H must be entered and its Required Action taken.

G.1 and G.2

Required Action G.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within similar ADS trip system Functions result in automatic initiation capability being lost for the ADS. Automatic initiation capability is lost if either (a) one Function 4.c channel and one Function 5.c channel are inoperable, (b) a combination of Function 4.e, 4.f, 5.e, and 5.f channels are inoperable such that channels associated with five or more low pressure ECCS pumps are inoperable, or (c) one or more Function 4.g channels and one or more Function 5.g channels are inoperable. 3

In this situation (loss of automatic initiation capability), the 96 hour or 8 day allowance, as applicable, of Required Action G.2 is not appropriate, and all ADS valves must be declared inoperable within 1 hour after discovery of loss of ADS initiation capability. The Note to Required Action G.1 states that Required Action G.1 is only applicable for Functions 4.c, 4.e, 4.f, 4.g, 5.c, 5.e, 5.f, and 5.g. Required Action G.1 is not applicable to Functions 4.h and 5.h (which also require entry into this Condition if a channel in these Functions is inoperable), since they are the Manual Initiation Functions and are not assumed in any accident or transient analysis. Thus, a total loss of manual initiation capability for 96 hours or 8 days (as allowed by Required Action G.2) is allowed. 3

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action G.1, the Completion Time only begins upon discovery that the ADS cannot be automatically initiated due to inoperable channels within similar ADS trip system Functions as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

BASES

ACTIONS (continued)

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time 3 of 8 days has been shown to be acceptable (Ref. ~~5~~) to permit restoration 1 of any inoperable channel to OPERABLE status if both HPCI and RCIC are OPERABLE (Required Action G.2). If either HPCI or RCIC is inoperable, the time shortens to 96 hours. If the status of HPCI or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCI or RCIC inoperability. However, the total time for an inoperable channel cannot exceed 8 days. If the status of HPCI or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable channel. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

H.1

With any Required Action and associated Completion Time not met, the associated feature(s) may be incapable of performing the intended function, and the supported feature(s) associated with inoperable untripped channels must be declared inoperable immediately.

SURVEILLANCE REQUIREMENTS

~~REVIEWER'S NOTE~~
~~Certain Frequencies are based on approved topical reports. In order for a licensee to use these Frequencies, the licensee must justify the Frequencies as required by the staff SER for the topical report.~~ 6

As noted in the beginning of the SRs, the SRs for each ECCS instrumentation Function are found in the SRs column of Table 3.3.5.1-1. The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may and be delayed for up to 6 hours as follows: (a) for Functions 3.c, 3.f~~, and 3.g~~; 3
and and (b) for Functions other than 3.c, 3.f~~, and 3.g~~ provided the associated Function or redundant Function maintains ECCS initiation capability. Upon completion of the Surveillance, or expiration of the 6 hour

BASES

SURVEILLANCE REQUIREMENTS (continued)

A channel that is shared between both trip systems is considered one channel.

allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is 3 1 based on the reliability analysis (Ref. 6) assumption of the average time 1 required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the ECCS will initiate when necessary.

SR 3.3.5.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK guarantees that undetected outright channel failure is limited to 12 hours; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.5.1.2 and SR 3.3.5.1.5 3

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. A 2 successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

for SR 3.3.5.1.2

The Frequency of 92 days is based on the reliability analyses of Reference 5. 3 3

INSERT 11

SR 3.3.5.1.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.5.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analyses. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than the setting accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 5. 3 1

7

, SR 3.3.5.1.6,

SR 3.3.5.1.4 and SR 3.3.5.1.5 3

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of SR 3.3.5.1.4 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. 7 3

INSERT 12

The Frequency of SR 3.3.5.1.5 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. 3 3

24

INSERT 13 3

3

INSERT 11

The Frequency of 12 months for SR 3.3.5.1.5 is based on the known reliability of the equipment and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

3

INSERT 12

The Frequency of SR 3.3.5.1.6 is based upon the assumption of a 12 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

3

INSERT 13

The SR 3.3.5.1.4 annotation in Table 3.3.5.1-1 for Functions 1.c, 1.d, 2.c, 2.d, 4.c, 4.d, 5.c, and 5.d has been modified by two Notes. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of instrument performance will verify that the instrument will continue to behave in accordance with design basis assumptions. The purpose of the assessment is to ensure confidence in the instrument performance prior to returning the instrument to service. These channels will also be identified in the Corrective Action Program. In accordance with procedures, entry into the Corrective Action Program will require review and documentation of the condition of OPERABILITY. The second Note requires the setting for the instrument be returned to within the as-left tolerance of the nominal trip setpoint. This will ensure that sufficient margin to the Safety Limit and /or Analytical Limit is maintained. If the setting for the instrument cannot be returned to within the as-left tolerance of the nominal trip setpoint, then the instrument channel shall be declared inoperable. The second Note also requires that the nominal trip setpoint and the methodology for calculating the as-left and the as-found tolerances be in a document controlled under 10 CFR 50.59 (i.e., Technical Requirements Manual (Ref. 4)).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1.~~6~~ 8 (3)

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.8.1, and LCO 3.8.2 overlaps this Surveillance to provide complete testing of the assumed safety function. (4)

The ~~18~~ 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the ~~18~~ 24 month Frequency. (3) (3)

SR 3.3.5.1.7

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in Reference 4.

ECCS RESPONSE TIME may be verified by actual response time measurements in any series of sequential, overlapping, or total channel measurements.

--------------------REVIEWER'S NOTE--------------------

[The following Bases are applicable for plants adopting NEDO-32291-A. (3)

However, the measurement of instrument loop response times may be excluded if the conditions of Reference 6 are satisfied.]

ECCS RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. The 18 month Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

BASES

REFERENCES

U

1. ~~F~~SAR, Section ~~[5.2]~~. 14.7.2 1 7

2. ~~FSAR, Section [6.3].~~ 1

U

2 ~~3~~. ~~F~~SAR, Chapter ~~[15]~~. 14 1 7

~~4. NEDC-31376-P, "Edwin I. Hatch Nuclear Power Plant, SAFER/GESTR-LOCA, Loss-of-Coolant Accident Analysis," December 1986.~~ 1

3 ~~5~~. NEDC-30936-P-A, "BWR Owners' Group Technical Specification s 1 and Improvement Analyses for ECCS Actuation Instrumentation, Part 2," December 1988.

~~[6. NEDO-32291-A, "System Analyses for the Elimination of Selected Response Time Testing Requirements," October 1995.]~~ 3

4. Technical Requirements Manual. 3

JUSTIFICATION FOR DEVIATIONS
ITS 3.3.5.1 BASES, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

1. Changes are made (additions, deletions, and/or changes) to the ISTS Bases, which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.
2. Editorial changes made for enhanced clarity or to be consistent with similar statements in other places in the Bases.
3. Changes are made to reflect changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.
4. Typographical/Grammatical error corrected.
5. Changes are made to reflect the Specification.
6. This Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.
7. The brackets have been removed and the proper plant specific information/value has been provided.

Specific No Significant Hazards Considerations (NSHCs)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

There are no specific NSHC discussions for this Specification.