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U-602196
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8E.100a

October 26, 1993

Docket No. 50-461

10CFR50.90

Document Control Desk
Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Clinton Power Station Proposed Amendment of Facility Operating License No. NPF-62

Dear Sir:

Pursuant to 10CFR50.90, Illinois Power (IP) hereby applies for amendment of Facility Operating License No. NPF-62, Appendix A - Technical Specifications, for Clinton Power Station (CPS). This request consists of proposed changes to improve and reformat the CPS Technical Specifications consistent with NUREG-1434, "Improved BWR-6 Technical Specifications", Revision 0, September 1992.

This submittal represents a significant effort on the part of the BWR-6/Mark III community to prepare consistent license amendment applications. As the BWR-6 owners have previously discussed with Mr. Grimes and others of your staff, implementation of this request will result in a significant contribution to increased safety in operation of CPS. In addition, this request implements cost saving efforts on the part of both the NRC and IP in that significant NRC resource savings can be realized through generic review of the application of NUREG-1434 to the BWR-6 plants. Further, implementation of this request reduces the likelihood of future Technical Specification changes for CPS by reducing IP's backlog of individual line item changes.

As part of the development of this request, IP has applied the criteria contained in the Final NRC Policy Statement on Technical Specification Improvements to the current CPS Technical Specifications utilizing BWR Owners' Group (BWROG) report NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," (and Supplement 1) as incorporated in NUREG-1434. The results of the application of the screening criteria for CPS are contained in Enclosure 1 to this letter.

Enclosure 2 to this letter is arranged by Technical Specification section. For each section two attachments are provided. Attachment 1 contains a markup of the current CPS Technical Specifications reflecting the changes necessary to adopt NUREG-1434 for CPS. Each of the changes is annotated to identify the applicable justification for the change. In addition, No Significant Hazards Consideration (NSHC) evaluations are

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provided for each of the changes. Generic NSHC evaluations have been provided at the beginning of Enclosure 2 for the changes annotated as "A" (administrative); "R" (relocated as identified in Enclosure 1); "M" (technical change - more restrictive; and "LA", "LB", and "LC" (generic technical change - less restrictive). The justifications and the NSHC evaluations for those changes annotated as "L" (specific technical change - less restrictive) are provided at the end of the associated Attachment 1.

Attachment 2 to Enclosure 2 of this letter contains a markup of NUREG-1434 to reflect the changes necessary to apply the NUREG to CPS. Each of the changes are annotated to identify the reasons for the change. Similar to Attachment 1 to Enclosure 2, each of the changes has been categorized for review purposes. The changes are identified as "B" (provides plant-specific information contained in brackets in NUREG-1434), "P" (provides plant specific information not contained in brackets in NUREG-1434), or "C" (denoting a generic change to NUREG-1434). The discussion/justification for each change to NUREG-1434 is contained at the end of the associated Attachment 2.

An affidavit supporting the facts set forth in this letter and its enclosures is provided in the attachment to this letter.

IP has reviewed the proposed changes against the criteria of 10CFR51.22 for categorical exclusion from environmental impact considerations. The proposed changes do not involve a significant hazards consideration, or significantly increase the amounts or change the types of effluents that may be released offsite, nor do they significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, IP concludes that the proposed changes meet the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirement for an Environmental Impact Statement.

In closing, this request represents expenditure of significant resources on the part of the BWR-6 utilities and culminates an accomplishment that sets a precedence in the industry. IP looks forward to working with your staff in obtaining your approval of this request.

Sincerely yours,

J. S. Perry
Senior Vice President

DAS/nls

Attachment
Enclosures

cc: NRC Clinton Licensing Project Manager
NRC Resident Office, V-690
Regional Administrator, Region III, USNRC
Illinois Department of Nuclear Safety

Attachment
to U-602196

STATE OF ILLINOIS
COUNTY OF DEWITT

J. Stephen Perry, being first duly sworn, deposes and says: That he is Senior Vice President of Illinois Power Company; that the application for amendment of Facility Operating License NPF-62 has been prepared under his supervision and direction; that he knows the contents thereof; and that to the best of his knowledge and belief said application and the facts contained therein are true and correct.

DATE: This 26 day of October 1993

Signed: J. Stephen Perry

J. Stephen Perry

Subscribed and sworn to before me this 26th day of October 1993.

Linda S. French
Notary Public

"OFFICIAL SEAL"
Linda S. French
Notary Public, State of Illinois
My Commission Expires 9/1/96

APPLICATION OF SELECTION CRITERIA TO THE

CLINTON POWER STATION

TECHNICAL SPECIFICATIONS

CONTENTS

1. INTRODUCTION

The purpose of this document is to confirm the results of the BWR Owners' Group (BWROG) application of the Technical Specification selection criteria on a plant specific basis for Clinton Power Station (CPS). Illinois Power Company (IP) has applied the selection criteria to each of the current CPS Technical Specifications utilizing the BWROG report NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment" (and Supplement 1), as incorporated in NUREG-1434, "BWR-6 Improved Technical Specifications," Revision 0. Additionally, in accordance with the NRC guidance, this confirmation of the application of selection criteria to CPS includes confirming the risk insights from Probabilistic Risk Assessment (PRA) evaluations, provided in the NEDO documents, as applicable to CPS.

2. SELECTION CRITERIA

IP has utilized the selection criteria provided in the NRC Final Policy Statement on Technical Specification Improvements of July 16, 1993, to develop the results contained in the attached matrix. Probabilistic Risk Assessment (PRA) insights as used in the BWROG submittal were utilized, confirmed by IP, and are discussed in the next section of this report. The selection criteria and discussion provided in the NRC Final Policy Statement 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.

Criterion 2: A process variable 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. These analyses consist of postulated events, analyzed in the Updated Safety Analysis Report (USAR), for which a structure, system, or component must meet specified functional goals. These analyses are contained in Chapters 6 and 15 of the USAR (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.

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. So long as these variables are maintained within the established values, risk to the public safety is presumed to be acceptably low.

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 Updated Safety Analysis Report (or equivalent chapters). Such a safety sequence analysis considers all applicable events, whether explicitly or implicitly presented. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria), so that the plant response to Design Basis Accidents and Transients limits the consequences of these events to within the appropriate acceptance criteria.

It is the intent of this criterion to capture into Technical Specifications only those structures, systems, and components that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function.

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

Discussion of Criterion 4: It is the Commission's policy that licensees retain in their Technical Specifications LCOs, action statements and Surveillance Requirements for the following systems (as applicable) which operating experience and PSA have generally shown to be significant to public health and safety, as well as any other structures, systems, or components that meet this criterion:

- Reactor Core Isolation Cooling (RCIC)/Isolation Condenser;
- Residual Heat Removal (RHR);
- Standby Liquid Control (SLC); and
- Recirculation Pump Trip (RPT).

The Commission recognizes that other structures, systems, or components may meet this criterion. Plant- and design-specific PSAs 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 which 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 Specifications-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. Furthermore, as part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to better use risk and reliability information to define future generic Technical Specification requirements.

3. PROBABILISTIC RISK ASSESSMENT (PRA) INSIGHTS

Introduction and Objectives

The Final Policy Statement includes a statement that the NRC expects Owners Groups to utilize the available literature on risk insights 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 the 10 CFR 50.59 review program. These Specifications have been compared to a variety of PRA material with two purposes: (1) to identify if a component or variable is addressed by PRA, and 2) if addressed, to judge if the component or variable is risk-important. In addition, in some cases risk was judged independent of any specific PRA material. The intent of the review was to provide a supplemental screen to the deterministic criteria. Technical Specifications to be retained were not reviewed. This review was accomplished in BWROG submittal NEDO-31466 and Supplement 1, except where discussed in Appendix A, "Justification For Specification Relocation," and has been confirmed by IP for those Specifications to be relocated.

Assumptions and Approach

Briefly, the approach used in NEDO-31466 and Supplement 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. The table provided at the end of this section lists the

PRAs used to provide insights for making the assessments. 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. LCOs which did not meet the screening criteria were evaluated. Those satisfying a criterion were not. 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 probability (high, medium, or 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 probability.

e. Determine the relative significance (high, medium, or 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.

BWR PRAs USED IN NEDO-31466
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, Revision 1, February 1986.

- 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 Activation Instrumentation) Part 2," June 1987.

4. RESULTS OF APPLICATION OF SELECTION CRITERIA

The selection criteria from Section 2 were applied to the CPS Technical Specifications. The following 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. 10 CFR 50.92 evaluations for those Specifications relocated are provided with the Discussion of Changes for the specific Technical Specifications. IP will relocate those Specifications identified as not satisfying the criteria in a dedicated section of the CPS USAR, programs, procedures, or other licensee controlled documents.

SUMMARY DISPOSITION MATRIX

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
1.0	DEFINITIONS	1.1 3.10.2 3.10.3 3.10.4	Yes	See Note 1 and Note 4.
2.1	SAFETY LIMITS	2.0		
2.1.1	THERMAL POWER, Low Pressure or Low Flow	2.1.1.1	Yes	See Note 2.
2.1.2	THERMAL POWER, High Pressure and High Flow	2.1.1.2	Yes	See Note 2.
2.1.3	Reactor Coolant System Pressure	2.1.2	Yes	See Note 2.
2.1.4	Reactor Vessel Water level	2.1.1.3	Yes	See Note 2.
2.2	LIMITING SAFETY SYSTEM SETTING			
2.2.1	Reactor Protection System (RPS) Instrumentation Setpoints	3.3.1.1	Yes	The application of Technical Specification selection criteria is not appropriate. However, the RPS LSSS have been included as part of the RPS Instrumentation Specification, which has been retained since the Functions either actuate to mitigate consequences of design basis accidents and transients or are retained as directed by the NRC as the Functions are part of the RPS.
3.0	LIMITING CONDITIONS FOR OPERATION - APPLICABILITY			
3.0.1	Operational Conditions	LCO 3.0.1	Yes	See Note 3.
3.0.2	Noncompliance	LCO 3.0.2	Yes	See Note 3.
3.0.3	Generic Actions	LCO 3.0.3	Yes	See Note 3.
3.0.4	Entry into Operational Conditions	LCO 3.0.4	Yes	See Note 3.
4.0	SURVEILLANCE REQUIREMENTS - APPLICABILITY			
4.0.1	Operational Conditions	SR 3.0.1	Yes	See Note 3.

SUMMARY DISPOSITION MATRIX

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
4.0.2	Time of Performance	SR 3.0.2	Yes	See Note 3.
4.0.3	Noncompliance	SR 3.0.3	Yes	See Note 3.
4.0.4	Entry into Operational Conditions	SR 3.0.4	Yes	See Note 3.
4.0.5	ASME Code Class 1, 2, 3 Components	5.7.10 5.7.11	Yes	See Note 3.
3/4.1	REACTIVITY CONTROL SYSTEM	3.1		
3/4.1.1	Shutdown Margin	3.1.1	Yes-2	Not a measured process variable, but is important parameter that is used to confirm the acceptability of the accident analysis.
3/4.1.2	Reactivity Anomalies	3.1.2	Yes-2	Confirms assumptions made in the reload safety analysis.
3/4.1.3	Control Rods			
3/4.1.3.1	Control Rod OPERABILITY	3.1.3 3.1.8	Yes-3	Primary success path in mitigating the consequences of design basis accidents and transients.
3/4.1.3.2	Control Rod Maximum Scram Insertion Times	3.1.3 3.1.4	Yes-3	Same as above.
3/4.1.3.3	Control Rod Scram Accumulators	3.1.5 3.9.5	Yes-3	Same as above.
3/4.1.3.4	Control Rod Drive Coupling	3.1.4	Yes-3	Same as above.
3/4.1.3.5	Control Rod Position Indication	3.1.3 3.9.4	Yes-3	Same as above.
3/4.1.3.6	Control Rod Drive Housing Support	Relocated	No	See Appendix A.
3/4.1.4	Control Rod Program Controls			
3/4.1.4.1	Control Rod Withdrawal	3.3.2.1	Yes-3	Prevents withdrawal of control rods that might exceed rod withdrawal error transient analysis assumptions.
3/4.1.4.2	Rod Pattern Control System	3.3.2.1 3.1.3 3.1.6	Yes-3	Prevents withdrawal of out-of-sequence control rods that might set up high rod worth conditions beyond CRDA assumptions. Also prevents deviation beyond banked position withdrawal sequence that if violated could allow high rod worth conditions that would challenge the MCPR Safety Limit and 1 percent cladding plastic strain fuel design limit during a rod withdrawal error event.
3/4.1.5	Standby Liquid Control System	3.1.7	Yes-4	Retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements due to potential risk significance.

SUMMARY DISPOSITION MATRIX

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
3/4.2	POWER DISTRIBUTION LIMITS	3.2		
3/4.2.1	Average Planar Linear Heat Generation Rate (APLHGR)	3.2.1	Yes-2	Peak cladding temperature following a LOCA is primarily dependent on initial APLHGR. As such, it is an initial condition of a DBA analysis.
3/4.2.2	Deleted in Amendment No. 18			
3/4.2.3	Minimum Critical Power Ratio	3.2.2	Yes-2	Utilized as an initial condition of the design basis analyses. DBA analysis are performed to establish the largest reduction in Critical Power Ratio. This value is added to the fuel cladding integrity safety limit to determine the MCPR value.
3/4.2.4	Linear Heat Generation Rate (LHGR)	3.2.3	Yes-2	LHGR is calculated to avoid exceeding plastic strain limits on fuel rods. As such, it is an initial condition of Design Basis Transient Analyses.
3/4.3	INSTRUMENTATION	3.3		
3/4.3.1	Reactor Protection System Instrumentation	3.3.1.1		
3/4.3.1.1	Intermediate Range Monitors	3.3.1.1	Yes	Retained as directed by the NRC, as it is part of the RPS System.
3/4.3.1.2	Average Power Range Monitors	3.3.1.1	Yes-3	Actuates to mitigate consequences of design basis accident or transient.
3/4.3.1.3	Reactor Vessel Steam Dome Pressure-High	3.3.1.1	Yes	Retained as directed by the NRC, as it is part of the RPS System.
3/4.3.1.4	Reactor Vessel Water Level-Low, Level 3	3.3.1.1	Yes-3	Actuates to mitigate consequences of design basis accident or transient.
3/4.3.1.5	Reactor Vessel Water Level-High, Level 8	3.3.1.1	Yes-3	Actuates to mitigate consequences of design basis accident or transient.
3/4.3.1.6	Main Steam Line Isolation Valve-Closure	3.3.1.1	Yes	Retained as directed by the NRC, as it is part of the RPS System.
3/4.3.1.7	Main Steam Line Radiation-High	Deleted	No	Deleted; see RPS technical change for MSLRM.
3/4.3.1.8	Drywell Pressure-High	3.3.1.1	Yes	Retained as directed by the NRC, as it is part of the RPS System.
3/4.3.1.9	Scram Discharge Volume Water Level-High	3.3.1.1	Yes	Retained as directed by the NRC, as it is part of the RPS System.
3/4.3.1.10	Turbine Stop Valve-Closure	3.3.1.1	Yes-3	Actuates to mitigate consequences of design basis accident or transient.

SUMMARY DISPOSITION MATRIX

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
3/4.3.1.11	Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure-Low	3.3.1.1	Yes-3	Actuates to mitigate consequences of design basis accident or transient.
3/4.3.1.12	Reactor Mode Switch Shutdown Position	3.3.1.1	Yes	Retained as directed by the NRC, as it is part of the RPS System.
3/4.3.1.13	Manual Scram	3.3.1.1	Yes	Retained as directed by the NRC, as it is part of the RPS System.
3/4.3.2	Containment and Reactor Vessel Isolation Control System	3.3.6.1 3.3.6.2		
3/4.3.2.1	Primary and Secondary Containment Isolation	3.3.6.1 3.3.6.2	Yes-3	Actuates to mitigate the consequences of a DBA LOCA.
3/4.3.2.1.m	Main Steam Line Radiation-High	Deleted	No	Deleted; see RPS technical change for MSLRM.
3/4.3.2.1.n	Manual Initiation	3.3.6.1 3.3.6.2	Yes	Retained as directed by the NRC, as it is part of the Isolation System.
3/4.3.2.2	Main Steam Line Isolation	3.3.6.1	Yes-3*	Actuates to mitigate the consequences of a DBA LOCA.
3/4.3.2.2.b	Main Steam Line Radiation-High	Deleted	No	Deleted; see RPS technical change for MSLRM.
3/4.3.2.2.i	Manual Initiation	3.3.6.1	Yes	Retained as directed by the NRC, as it is part of the Isolation System.
3/4.3.2.3	Reactor Water Cleanup System Isolation	3.3.6.1	Yes-3*	Actuates to isolate potential leakage paths to secondary containment consistent with safety analysis assumptions.
3/4.3.2.3.h	SLCS Initiation	3.3.6.1	Yes-4	Retained due to importance of SLCS and in accordance with the NRC Final Policy Statement on Technical Specification Improvements.
3/4.3.2.3.i	Manual Initiation	3.3.6.1	Yes	Retained as directed by the NRC, as it is part of the Isolation System.
3/4.3.2.4	Reactor Core Isolation Cooling System Isolation	3.3.6.1	Yes-3*	Actuates to isolate potential leakage paths to secondary containment consistent with safety analysis assumptions.
3/4.3.2.4.c	RCIC Steam Supply Pressure-Low	3.3.6.1	Yes	Does not satisfy the selection criteria; however, it is being retained due to potential risk significance.
3/4.3.2.4.d	RCIC Turbine Exhaust Diaphragm Pressure-High	3.3.6.1	Yes	Does not satisfy the selection criteria; however, it is being retained due to potential risk significance.
3/4.3.2.4.k	Manual Initiation	3.3.6.1	Yes	Retained as directed by the NRC, as it is part of the Isolation System.

* Except Ambient Temperature and Differential Temperature Instruments, which are to be Relocated. See Appendix A.

SUMMARY DISPOSITION MATRIX

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
3/4.3.2.5	RHR System Isolation	3.3.6.1	Yes-3*	Actuates to isolate potential leakage paths to secondary containment consistent with safety analysis assumptions.
3/4.3.2.5.c	Reactor Vessel Water Level-Low, Level 3	3.3.6.1	Yes	Retained due to importance of RHR System and the NRC Final Policy Statement on Technical Specification Improvements.
3/4.3.2.5.e	Reactor Vessel (RHR Cut-in Permissive) Pressure-High	3.3.6.1	Yes-4	Retained due to importance of the RHR System and in accordance with the NRC Final Policy Statement on Technical Specification Improvements.
3/4.3.2.5.g	Manual Initiation	3.3.6.1	Yes	Retained as required by the NRC, as it is part of the Primary Containment Isolation System.
3/4.3.3	Emergency Core Cooling System Actuation Instrumentation	3.3.5.1 3.3.8.1		
3/4.3.3.A	Division 1 Trip System	3.3.5.1	Yes-3	ECCS mitigate the consequences of a DBA LOCA.
3/4.3.3.A.1.g	Manual Initiation	3.3.5.1	Yes	Retained as required by the NRC, as it is part of the ECCS Actuation System.
3/4.3.3.A.2.h	Manual Inhibit ADS Switch	Relocated	No	See Appendix A.
3/4.3.3.A.2.i	Manual Initiation	3.3.5.1	Yes	Retained as required by the NRC, as it is part of the ECCS Actuation System.
3/4.3.3.B	Division II Trip System	3.3.5.1	Yes-3	ECCS mitigate the consequences of a DBA LOCA.
3/4.3.3.B.1.g	Manual Initiation	3.3.5.1	Yes	Retained as required by the NRC, as it is part of the ECCS Actuation System.
3/4.3.3.B.2.g	Manual Inhibit ADS Switch	Relocated	No	See Appendix A.
3/4.3.3.B.2.h	Manual Initiation	3.3.5.1	Yes	Retained as required by the NRC, as it is part of the ECCS Actuation System.
3/4.3.3.C	Division III Trip System	3.3.5.1	Yes-3	ECCS mitigate the consequences of a DBA LOCA.
3/4.3.3.C.1.c	Reactor Vessel Water Level-High, Level 8	3.3.5.1	Yes	Does not satisfy the selection criteria; however, it is being retained due to potential risk significance.
3/4.3.3.C.1.h	Manual Initiation	3.3.5.1	Yes	Retained as required by the NRC, as it is part of the ECCS Actuation System.
3/4.3.3.D	Loss of Power	3.3.8.1	Yes-3	Loss of power instrumentation actuates to assure power availability to the ECCS in the event of a loss of offsite power. Mitigation of DBAs relies on the availability of the ECCS and ECCS power supply.

*Except Ambient Temperature and Differential Temperature Instruments, which are to be Relocated. See Appendix A.

SUMMARY DISPOSITION MATRIX

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
3/4.3.4	Recirculation Pump Trip Actuation Instrumentation			
3/4.3.4.1	ATWS Recirculation Pump Trip System Instrumentation	3.3.4.2	Yes-4	Retained due to importance of ATWS Recirculation Pump Trip System and in accordance with the NRC Final Policy Statement on Technical Specification Improvements.
3/4.3.4.2	End-of-Cycle Recirculation Pump Trip System Instrumentation	3.3.4.1	Yes-3	EOC-RPT aids the reactor scram in protecting fuel cladding integrity by ensuring the fuel cladding integrity safety limit is not exceeded during a load rejection or turbine trip transient.
3/4.3.5	Reactor Core Isolation Cooling (RCIC) System Actuation Instrumentation	3.3.5.2	Yes-3&4	Required to mitigate the consequences of a DBA and retained due to importance of the RCIC System and in accordance with the NRC Final Policy Statement on Technical Specification Improvements.
3/4.3.5.b	Reactor Vessel Water Level-High, Level 8	3.3.5.2	Yes	Does not satisfy the selection criteria; however, it is being retained due to potential risk significance.
3/4.3.5.e	Manual Initiation	3.3.5.2	Yes	Retained as required by the NRC, as it is part of the RCIC System Actuation System.
3/4.3.6	Control Rod Block Instrumentation	3.3.2.1		
3/4.3.6.1	Rod Pattern Control System	3.3.2.1	Yes-3	Prevents withdrawal of out-of-sequence control rods that might set up high rod worth conditions beyond CRDA assumptions. Also prevents deviation beyond a banked position withdrawal sequence that if violated could allow high rod worth conditions that would challenge the MCPR Safety Limit and 1 percent cladding plastic strain fuel design limit during a rod withdrawal error event.
3/4.3.6.2	APRM	Relocated	No	See Appendix A.
3/4.3.6.3	Source Range Monitors	Relocated	No	See Appendix A.
3/4.3.6.4	Intermediate Range Monitors	Relocated	No	See Appendix A.
3/4.3.6.5	Scram Discharge Volume	Relocated	No	See Appendix A.
3/4.3.6.6	Reactor Coolant System Recirculation Flow	Relocated	No	See Appendix A.
3/4.3.6.7	Reactor Mode Switch	3.3.2.1 3.9.2	Yes-3	Prevents control rods from being withdrawn when shutdown, thus ensuring that inadvertent rod withdrawal error cannot occur during MODES 3 and 4. Ensures one-rod-out interlock is enforced.

SUMMARY DISPOSITION MATRIX

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
3/4.3.7	Monitoring Instrumentation			
3/4.3.7.1	Radiation Monitoring Instrumentation	3.3.7.1		
3/4.3.7.1.1	Main Control Room Air Intake Radiation Monitor	3.3.7.1	Yes-3	Actuates to maintain control room habitability so that
3/4.3.7.1.2	Area Monitors	Relocated	No	See Appendix A.
3/4.3.7.2	Seismic Monitoring Instrumentation	Relocated	No	See Appendix A.
3/4.3.7.3	Meteorological Monitoring Instrumentation	Relocated	No	See Appendix A.
3/4.3.7.4	Remote Shutdown Monitoring Instrumentation	3.3.3.2	Yes	Does not satisfy the selection criteria; however, it is being retained as directed by the NRC as a significant contributor to risk reduction.
3/4.3.7.5	Accident Monitoring Instrumentation	3.3.3.1	Yes-3	See Appendix A.
3/4.3.7.6	Source Range Monitors	3.3.1.2	Yes	Does not satisfy the selection criteria; however, it is being retained because the NRC considers it necessary for flux monitoring during shutdown, startup and refueling operations.
3/4.3.7.7	Traversing In-Core Probe System	Relocated	No	See Appendix A.
3/4.3.7.8	Chlorine Detection System	Relocated	No	See Appendix A.
3/4.3.7.9	Removed by Previous Amendment			
3/4.3.7.10	Loose-Part Detection System	Relocated	No	See Appendix A.
3/4.3.7.11	Main Condenser Offgas Treatment System Explosive Gas Monitoring Instrumentation	Relocated	No	See Appendix A.
3/4.3.8	Removed in Amendment No. 60.			
3/4.3.9	Plant Systems Actuation Instrumentation	3.3.6.3 3.3.6.4		
3/4.3.9.1	Containment Spray System	3.3.6.3	Yes-3	Actuates to mitigate consequences of DBA LOCA.
3/4.3.9.1.e	Manual Initiation	3.3.6.3	Yes	Retained as required by the NRC, as it is part of the Containment Spray System Actuation System.
3/4.3.9.2	Feedwater System/Main Turbine Trip System	Relocated	No	See Appendix A.
3/4.3.9.3	Suppression Pool Makeup System	3.3.6.4	Yes-3	Actuates to mitigate consequences of DBA LOCA.
3/4.3.9.3.e	SPMS Manual Initiation	3.3.6.4	Yes	Retained as required by the NRC, as it is part of the SPMS System Actuation System.
3/4.3.9.3.f	SPMS Mode Switch Permissive	Relocated	No	See Appendix A.
3/4.3.10	Nuclear Systems Protection System-Self Test System	Relocated	No	See Appendix A.

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
3/4.4	REACTOR COOLANT SYSTEM	3.4		
3/4.4.1	Recirculation System			
3/4.4.1.1	Recirculation Loops	3.4.1 3.4.2 3.4.11	Yes-2	Recirculation loop flow is an initial condition in the safety analysis. Closure of the flow control valves within specified time limits functions to mitigate the consequences of a LOCA.
3/4.4.1.2	Jet Pumps	3.4.3	Yes-2	Jet pump OPERABILITY is assumed in the LOCA analyses to assure adequate core reflood capability.
3/4.4.1.3	Recirculation Loop Flow	3.4.1	Yes-2	Recirculation loop flow mismatch within limits is an initial condition in the safety analysis.
3/4.4.1.4	Idle Recirculation Loop Startup	3.4.11	Yes-2	Temperature differential between the reactor coolant in the reactor vessel and the idle loop is an initial condition in the transient analysis. Idle loop startup with temperatures outside the limit could result in a reactivity transient and potential violation of the Safety Limit MCPR.
3/4.4.2	Safety Valves			
3/4.4.2.1	Safety/Relief Valves	3.3.6.5 3.4.4	Yes-3	A minimum number of Safety/Relief Valves is assumed in the safety analyses to mitigate overpressure events.
3/4.4.2.2	Safety/Relief Valves Low-Low Set Function	3.3.6.5 3.6.1.6	Yes-3	A minimum number of Safety/Relief Valves is assumed in the containment loading safety analysis.
3/4.4.3	Reactor Coolant System Leakage			
3/4.4.3.1	Leakage Detection Systems	3.4.7	Yes-1	Leak detection is used to indicate a significant abnormal condition of the reactor coolant pressure boundary.
3/4.4.3.2	Operational Leakage	3.4.5 3.4.6	Yes-2	Leakage beyond limits would indicate a significant abnormal condition of the reactor coolant pressure boundary. Operation in this condition is unanalyzed and may result in reactor coolant pressure boundary failure.
3/4.4.4	Chemistry	Relocated	No	See Appendix A.
3/4.4.5	Specific Activity	3.4.8	Yes-2	Specific activity provides an indication of the onset of significant fuel cladding failure and is a variable used in the DBA analysis.
3/4.4.6	Reactor Coolant System			

SUMMARY DISPOSITION MATRIX

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
3/4.4.6.1	Pressure/Temperature Limits	3.4.11 5.8.1.7	Yes-2	This LCO establishes initial conditions for operation such that operation is prohibited in areas or at temperature rate changes that might cause undetected flaws to propagate, in turn challenging the reactor coolant pressure boundary integrity.
3/4.4.6.2	Reactor Steam Dome	3.4.12	Yes-2	The reactor steam dome pressure is an initial condition for the overpressurization analyses.
3/4.4.7	Main Steam Line Isolation Valves	3.6.1.3	Yes-3	Main steam line isolation within specified time limits ensures that the release to the environment is consistent with the assumptions in the LOCA analysis.
3/4.4.8	Structural Integrity	Relocated	No	See Appendix A.
3/4.4.9	Residual Heat Removal			
3/4.4.9.1	Hot Shutdown	3.4.9 3.10.1	Yes-4	Retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements due to potential risk significance.
3/4.4.9.2	Cold Shutdown	3.4.10 3.10.1	Yes-4	Same as above
3/4.5	EMERGENCY CORE COOLING SYSTEMS	3.5		
3/4.5.1	ECCS-Operating	3.5.1 5.8.2	Yes-3	Functions to mitigate the consequences of a DBA.
3/4.5.2	ECCS-Shutdown	3.5.2	Yes-3	Functions to mitigate the consequences of a vessel draindown event.
3/4.5.3	Suppression Pool	3.5.2 3.6.2.2	Yes-2&3	Functions to mitigate the consequences of a DBA and a vessel draindown event.
3/4.6	CONTAINMENT SYSTEMS	3.6		
3/4.6.1	Primary Containment			
3/4.6.1.1	Primary Containment Integrity	3.6.1.1 3.6.1.3	Yes-2&3	Containment integrity functions to mitigate the consequences of a DBA.
3/4.6.1.2	Primary Containment Leakage	3.6.1.1 3.6.1.2 3.6.1.3	Yes	This LCO does not satisfy the selection criteria; however, containment leakage is an assumption utilized in the LOCA safety analysis (but it is not a process variable). Therefore, it is being retained as a Surveillance Requirement.

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
3/4.6.1.3	Primary Containment Air Locks	3.6.1.2	Yes-3	Credit for air tightness is considered in safety analysis to limit offsite dose rates during a DBA.
3/4.6.1.4	MSIV Leakage Control System	3.6.1.8	Yes-3	Assumed in primary containment isolation events to direct the release of untreated leakage from the MSIVs such that offsite dose is within 10 CFR 100 guidelines.
3/4.6.1.5	Containment Structural Integrity	3.6.1.1	Yes-3	Containment functions to mitigate the consequences of a DBA.
3/4.6.1.6	Containment Internal Pressure	3.6.1.4	Yes-2	Containment pressure is an initial condition in the LOCA safety analysis.
3/4.6.1.7	Primary Containment Average Air Temperature	3.6.1.5	Yes-2	Primary containment air temperature is an initial condition in the LOCA safety analysis.
3/4.6.1.8	Containment Building Ventilation and Purge Systems	3.6.1.3	Yes-3	System is part of primary success path for an accident involving release of radioactivity offsite.
3/4.6.2	Drywell			
3/4.6.2.1	Drywell Integrity	3.6.5.1 3.6.5.3	Yes-2&3	Drywell integrity functions to mitigate the consequences of a DBA.
3/4.6.2.2	Drywell Bypass Leakage	3.6.5.1	Yes	This LCO does not satisfy the selection criteria; however, drywell bypass leakage is an assumption utilized in the LOCA safety analysis (but it is not a process variable). Therefore, it is being retained as a Surveillance Requirement.
3/4.6.2.3	Drywell Air Locks	3.6.5.2	Yes-3	Credit for drywell air lock leakage is an assumption utilized in the LOCA safety analysis.
3/4.6.2.4	Drywell Structural Integrity	3.6.5.1	Yes-3	Drywell functions to mitigate the consequences of a DBA.
3/4.6.2.5	Drywell Internal Pressure	3.6.5.4	Yes-2	Drywell pressure is an initial condition in the LOCA safety analysis.
3/4.6.2.6	Drywell Average Air Temperature	3.6.5.5	Yes-2	Drywell air temperature is an initial condition in the LOCA safety analysis.
3/4.6.2.7	Drywell Vent and Purge System	3.6.5.3	Yes-3	Isolation valves function to limit consequences of a DBA LOCA.
3/4.6.3	Depressurization Systems			
3/4.6.3.1	Suppression Pool	3.6.2.1 3.6.2.2	Yes-2&3	Suppression pool water level and temperature are initial conditions in the DBA LOCA analysis and mitigate the consequences of the DBA.
3/4.6.3.2	Containment Spray	3.6.1.7	Yes-3	Containment spray is assumed to mitigate the consequences of a DBA LOCA.

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
3/4.6.3.3	Suppression Pool Cooling	3.6.2.3	Yes-3	Suppression pool cooling functions to limit the effects of a DBA.
3/4.6.3.4	Suppression Pool Makeup System	3.6.2.4	Yes-3	SPMU System functions to mitigate the consequences of a DBA LOCA.
3/4.6.4	Primary Containment Isolation Valves	3.6.1.3	Yes-3	Isolation valves function to limit DBA consequences.
3/4.6.5	Drywell Post-LOCA Vacuum Relief Valves	3.6.5.6	Yes-3	Helps ensure drywell functions properly to mitigate the consequences of a DBA LOCA.
3/4.6.6	Secondary Containment			
3/4.6.6.1	Secondary Containment Integrity	3.6.4.1 3.6.4.2	Yes-3	Secondary containment integrity is relied on to limit the offsite dose during an accident by ensuring that any release to containment is delayed and treated before being released to the environment.
3/4.6.6.2	Secondary Containment Automatic Isolation Dampers	3.6.4.2	Yes-3	Valve operation within time limits establishes secondary containment and limits offsite dose releases to acceptable values.
3/4.6.6.3	Standby Gas Treatment System	3.6.4.3 5.7.12	Yes-3	Operation following a DBA acts to mitigate the consequences of offsite releases.
3/4.6.7	Atmosphere Control			
3/4.6.7.1	Containment Hydrogen Recombiner Systems	3.6.3.1	Yes-3	Operates post LOCA to limit hydrogen and oxygen concentrations to below explosive concentrations that might otherwise challenge containment integrity.
3/4.6.7.2	Containment/Drywell Hydrogen Mixing System	3.6.3.3	Yes-3	Operates post LOCA to limit hydrogen and oxygen concentrations to below explosive concentrations that might otherwise challenge containment integrity.
3/4.6.7.3	Primary Containment/Drywell Hydrogen Ignition System	3.6.3.2	Yes	While this Specification does not meet any selection criteria, it is being retained as directed by the NRC. Operates post LOCA to limit hydrogen and oxygen concentrations to below explosive concentrations that might otherwise challenge containment integrity.
3/4.7	PLANT SYSTEMS	3.7		
3/4.7.1	Service Water Systems			
3/4.7.1.1	Shutdown Service Water System (Loops A, B, C)	3.7.1 3.7.2	Yes-3	Designed for heat removal for safety related systems following a DBA. As such, acts to mitigate the consequences of an accident.

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
3/4.7.1.2	Ultimate Heat Sink	3.7.1	Yes-3	Heat sink for heat removal from safety related systems following a DBA. As such, acts to mitigate the consequences of an accident.
3/4.7.2	Control Room Ventilation System	3.7.3 3.7.4 5.7.12	Yes-3	Mitigates the consequences of an accident by maintaining habitability of the control room so that operators can remain in the control room following an accident and continue accident mitigation activities therefrom. It also functions to mitigate the consequences of an accident by ensuring that control room temperature is maintained such that control room safety related equipment remains OPERABLE following an accident.
3/4.7.3	Reactor Core Isolation Cooling System	3.5.3 3.3.5.2	Yes-3&4	Required to mitigate the consequences of a DBA and retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements due to potential risk significance.
3/4.7.4	Snubbers	Relocated	No	See Appendix A.
3/4.7.5	Sealed Source Contamination	Relocated	No	See Appendix A.
3/4.7.6	Main Turbine Bypass System	3.7.6	Yes-3	Actuates to mitigate the consequences of a feedwater controller failure-maximum demand transient and a turbine trip with bypass event.
3/4.7.7	Liquid Storage Tanks	5.7.13	Yes	Does not satisfy selection criteria; however, it is retained as a program in Administrative Controls as directed by the NRC.
3/4.7.8	Main Condenser Offgas Monitoring			
3/4.7.8.1	Offgas-Explosive Gas Mixture	5.7.13	Yes	Does not satisfy selection criteria; however, it is retained as a program in Administrative Controls as directed by the NRC.
3/4.7.8.2	Offgas-Noble Gas Radioactivity Rate	3.7.5	Yes-2	Main condenser offgas activity is an initial condition in the offgas system failure event.
3/4.8	ELECTRICAL POWER SYSTEMS	3.8		
3/4.8.1	AC Sources			
3/4.8.1.1	AC Sources-Operating	3.8.1 3.8.3 5.7.14 5.8.2	Yes-3	Required to mitigate the consequences of a LOA.
3/4.8.1.2	AC Sources-Shutdown	3.8.2 3.8.3 5.7.14 5.8.2	Yes-3	Functions to mitigate the consequences of a vessel draindown event and is needed to support NRC Final Policy Statement requirement for decay heat removal.

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
3/4.8.2	DC Sources			
3/4.8.2.1	DC Sources-Operating	3.8.4 3.8.6	Yes-3	Required to mitigate the consequences of a DBA.
3/4.8.2.2	DC Sources-Shutdown	3.8.5 3.8.6	Yes-3	Functions to mitigate the consequences of a vessel draindown event and is being retained to support the NRC Final Policy Statement requirement for decay heat removal.
3/4.8.3	Onsite Power Distribution Systems			
3/4.8.3.1	Distribution-Operating	3.8.7 3.8.9	Yes-3	Required to mitigate the consequences of a DBA.
3/4.8.3.2	Distribution-Shutdown	3.8.8 3.8.10	Yes-3	Functions to mitigate the consequences of a vessel draindown event and is being retained to support the NRC Final Policy Statement requirement for decay heat removal.
3/4.8.4	Electrical Equipment Protective Devices			
3/4.8.4.1	Containment Penetration Conductor Overcurrent Protective Devices	Relocated	No	See Appendix A.
3/4.8.4.2	Motor Operated Valves Thermal Overload Protection	Relocated	No	See Appendix A.
3/4.8.4.3	Reactor Protection System (RPS) Electric Power Monitoring	3.3.8.2	Yes-3	Provides protection for the RPS bus-powered instrumentation against unacceptable voltage and frequency conditions that could degrade the instrumentation so that it would not perform the intended safety function.
3/4.9	REFUELING OPERATIONS	3.9		
3/4.9.1	Reactor Mode Switch	3.9.1 3.9.2	Yes-3	Provides an interlock to preclude fuel loading with control rods withdrawn. Operation is assumed in the control rod removal error during refueling and fuel assembly insertion error during refueling accident analysis.
3/4.9.2	Instrumentation	3.3.1.2	Yes	Does not satisfy selection criteria; however, it is retained because the NRC considers it necessary for flux monitoring during shutdown, startup and refueling operations.
3/4.9.3	Control Rod Position	3.9.3	Yes-3	All control rods must be fully inserted when loading fuel. This requirement is assumed as an initial condition in the fuel assembly insertion error during refueling accident analysis.

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
3/4.9.4	Decay Time	Relocated	No	Although this LCO satisfied Criterion 2, the activities necessary prior to commencing movement of irradiated fuel ensure that there will always be 24 hours of subcriticality before movement of any irradiated fuel. Hence this Specification has been relocated.
3/4.9.5	Communications	Relocated	No	See Appendix A.
3/4.9.6	Fuel Handling Equipment			
3/4.9.6.1	Refueling Platform	Relocated	No	See Appendix A.
3/4.9.6.2	Auxiliary Platform	Relocated	No	See Appendix A.
3/4.9.7	Crane Travel-Spent Fuel Storage [illegible] Upper Containment Fuel Pool, and New [illegible] Storage Vault	Relocated	No	See Appendix A.
3/4.9.8	Water Level-Reactor Vessel	3.9.6 3.9.7	Yes-2	A minimum amount of water is required to assure adequate scrubbing of fission products following a fuel handling accident.
3/4.9.9	Water Level-Spent Fuel Storage and Upper Containment Fuel Pools	3.7.7	Yes-2	Same as above.
3/4.9.10	Control Rod Removal			
3/4.9.10.1	Single Control Rod Removal	3.10.4 3.10.5	Yes	See Note 4.
3/4.9.10.2	Multiple Control Rod Removal	3.10.6	Yes	See Note 4.
3/4.9.11	Residual Heat Removal and Coolant Circulation			
3/4.9.11.1	High Water Level	3.9.8	Yes-4	Retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements due to potential risk significance.
3/4.9.11.2	Low Water Level	3.9.9	Yes-4	Same as above.
3/4.9.12	Inclined Fuel Transfer System	Relocated	No	See Appendix A.
3/4.10	SPECIAL TEST EXCEPTIONS	3.10		
3/4.10.1	Primary Containment Integrity/Drywell Integrity	Deleted	No	The latitude of this Special Test Exception is not required at CPS.

SUMMARY DISPOSITION MATRIX

Old TS	Title	New Number	Retained/Criterion for Inclusion	Bases for Inclusion/Exclusion
3/4.10.2	Rod Pattern Control System	3.10.7	Yes	See Note 4.
3/4.10.3	Shutdown Margin Demonstrations	3.10.8	Yes	See Note 4.
3/4.10.4	Recirculation Loops	Deleted	No	The latitude of this Special Test Exception is not required at CPS.
3/4.10.5	Training Startups	3.10.9	Yes	See Note 4.
3/4.10.6	Special Instrumentation-Initial Core Loading	Deleted	No	This Specification is only allowed during initial core loading, which has been completed. Therefore, it is no longer applicable or needed and has been deleted.
5.0	DESIGN FEATURES	4.0	Yes	See Note 5.
6.0	ADMINISTRATIVE CONTROLS	5.0	Yes	See Note 6.

SUMMARY DISPOSITION MATRIX

NOTE 1: DEFINITIONS

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.

NOTE 2: SAFETY LIMITS/LSSS

Application of Technical Specification selection criteria is not appropriate. However, Safety Limits and Limiting Safety System Settings (as part of Reactor Protection System Instrumentation) will be included in Technical Specifications as required by 10 CFR 50.36.

NOTE 3: GENERIC 3.0/4.0

These Specifications provide generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operation and Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements of 3.0/4.0 will be retained in Technical Specifications. Human factors improvements and certain technical improvements have been implemented as agreed upon during NUMARC/NRC negotiations.

NOTE 4: SPECIAL TEST EXCEPTIONS

These Specifications are provided to allow relaxation of certain Limiting Conditions for Operation under certain specific conditions to allow testing and maintenance. They are directly related to one or more Limiting Conditions for Operation. Direct application of the Technical Specification selection criteria is not appropriate. However, those special test exceptions, directly tied to Limiting Conditions for Operation that remain in Technical Specifications, will also remain as Technical Specifications. Those special test exceptions not applicable at CPS have been deleted.

NOTE 5: DESIGN FEATURES

Application of Technical Specification selection criteria is not appropriate. However, Design Features will be included in Technical Specifications as required by 10 CFR 50.36.

NOTE 6: ADMINISTRATIVE CONTROLS

Application of Technical Specification selection criteria is not appropriate. However, Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.

APPENDIX A

JUSTIFICATION FOR

SPECIFICATION RELOCATION

3/4.1.3.6 CONTROL ROD DRIVE HOUSING SUPPORT

LCO Statement

The control rod drive housing support shall be in place.

Discussion:

Control rod drive housing support supports control rod OPERABILITY by plant configuration management. As such, control rod OPERABILITY cannot be satisfied without the support being in place. Without control rod OPERABILITY confirmed, appropriate Action Statements of the control rod OPERABILITY Specification must be entered. There is no need for duplicate requirements in a subsystem LCO. Relocation of this LCO is appropriate since plant configuration (the control rod housing support in place) would be controlled by post-maintenance procedures.

Comparison to Screening Criteria:

1. The control rod drive housing support is neither 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 control rod drive housing support is not capable of monitoring a process variable that is an initial condition of a DBA or transient analyses.

3. The control rod drive housing support is not part of a primary success path in the mitigation of a DBA or transient. It does support the Control Rod OPERABILITY Specification which has been retained in the CPS Technical Specifications. As such, having the control drive housing support not in place impacts control rod OPERABILITY, and appropriate actions are initiated which bound those actions that would be implemented because of the housing support being out of place. There is no need for duplicate actions. Control rod drive housing support contribution to DBA and transient mitigation is preserved by the Control Rod OPERABILITY Specification (LCO 3.1.3) requirements.

Probabilistic Risk Assessments (PRAs) address system risk contribution and identify systems which can be significant risk contributors to core damage and offsite releases. Subcomponent risk contribution (in this case, the control rod drive housing support) is not addressed as the inoperability of the supported system (control rod drive system) and subsequent risk contribution is bounding.

Conclusion:

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

3/4.3.2 CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM

LCO Statement:

The containment and reactor vessel isolation control system (CRVICS) channels shown in Table 3.3.2-1 shall be OPERABLE, with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

Ambient Temperature Isolation Instrumentation and Differential Temperature Isolation Instrumentation.

Discussion:

The ambient temperature and differential temperature instruments proposed to be relocated are not assumed to function to mitigate any accident described in Chapters 6 or 15 of the Updated Safety Analysis Report. These ambient and differential temperature instruments are provided only to detect and initiate isolation of a 25-gpm-equivalent steam leak. However, these instruments constitute only one method of determining steam leakage in their respective areas. In addition to the temperature monitoring, excess reactor coolant leakage can be detected by low reactor water level, high process line flow, high differential flow, and various other plant specific methods.

Comparison to Screening Criteria:

1. The Ambient Temperature and Differential Temperature Isolation Instruments are neither 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 Ambient Temperature and Differential Temperature Isolation Instruments are neither used for, nor capable of, monitoring a process variable that is an initial condition of a DBA or transient analyses.

3. The Ambient Temperature and Differential Temperature Isolation Instruments are not used as parts of a primary success path in the mitigation of a DBA or transient. No pressure-temperature analyses, radiation dose calculations, or equipment qualification parameters take credit for the operation of these ambient or differential temperature instruments. In addition, adequate redundancy is available to perform their functions by other methods.

Although the overall Isolation Instrumentation Function satisfies Criterion 3 of the NRC's Final Policy Statement on Technical Specification Improvement, these ambient temperature and differential temperature instruments are not assumed to function to mitigate any DBA or transient analyses.

Conclusion:

Since the screening criteria have not been satisfied, the ambient and differential temperature instrument functions requirements of the Improved Technical Specification Isolation Action Instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LCO Statement:

The Emergency Core Cooling System (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE, with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

3/4.3.3.A.2.h ADS Trip System 1—Manual Inhibit ADS Switch

3/4.3.3.B.2.g ADS Trip System 2—Manual Inhibit ADS Switch

Discussion:

The ADS Manual 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 activation would dilute sodium pentaborate injected by the Standby Liquid Control (SLC) System, thereby reducing the effectiveness of the SLC System shutdown.

Comparison to Screening Criteria:

1. The Manual Inhibit ADS Switch is neither 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 Manual Inhibit ADS Switch is neither used for, nor capable of, monitoring a process variable that is an initial condition of a DBA or transient analyses.

3. The Manual Inhibit ADS 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 mitigate the consequences of an ATWS event which is not a DBA or transient. The switch does support the ADS system, which has been retained in the CPS Technical Specifications. The actions to be taken in the event that this switch is positioned to defeat the ADS logic bound those actions to be taken if the switch is inoperable. There is no need for duplicate actions. The ADS actuation instrumentation requirements (LCO 3.3.5.1) preserve ADS Manual Inhibit Switch contribution to DBA and transient mitigation.

As discussed in Section 3.5 and summarized in Table 4-1 (item 112B) of NEDO-31466, the loss of the Manual Inhibit ADS Switch was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

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

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LCO Statement:

The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE, with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

3/4.3.6.2 APRM

Discussion:

The control rod block instrumentation is provided to prevent a control rod withdrawal error at power transient. APRMs utilize LPRM signals to provide information about the average core power. As such, they are not capable of providing the local power information necessary to mitigate a control rod withdrawal error transient. Therefore, these instruments are not used to mitigate a design basis accident (DBA) or transient.

Comparison to Screening Criteria:

1. The APRM control rod block is neither 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 analyses.

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

As discussed in Section 3.5 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. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the control rod block LCO and Surveillances applicable to APRM instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LCO Statement:

The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE, with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

3/4.3.6.3 Source Range Monitors

Discussion:

The control rod block instrumentation is provided to prevent a control rod withdrawal error at power transient. Source Range Monitor (SRM) signals are used to monitor neutron flux during refueling, shutdown, and startup conditions. 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 is neither 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 analyses.

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

As discussed in section 3.5 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 non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the control rod block LCO and Surveillances applicable to SRM instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LCO Statement:

The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE, with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

3/4.3.6.4 Intermediate Range Monitors

Discussion:

The control rod block instrumentation is provided to prevent a control rod withdrawal error at power transient. Intermediate Range Monitors (IRMs) are provided to monitor the neutron flux levels during refueling, shutdown, and startup conditions. 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 is neither 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 analyses.

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

As discussed in section 3.5 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. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the control rod block LCO and Surveillances applicable to IRM instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LCO Statement:

The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE, with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

3/4.3.6.5 Scram Discharge Volume

Discussion:

The control rod block instrumentation is provided to prevent a control rod withdrawal error at power transient. The purpose of measuring the scram discharge volume (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 analysis takes credit for rod block signals initiated by the SDV instrumentation.

Comparison to Screening Criteria:

1. The SDV control rod block is neither 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 analyses.

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

As discussed in section 3.5 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 non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the control rod block LCO and Surveillances applicable to SDV instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LCO Statement:

The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE, with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

3/4.3.6.6 Reactor Coolant System Recirculation Flow

Discussion:

The control rod block instrumentation is provided to prevent a control rod withdrawal error at power transient. Reactor recirculation flow provides input to the flow-biased setpoints of the APRMs. No design basis accident (DBA) or transient analysis takes credit for rod block signals initiated by the APRMs.

Comparison to Screening Criteria:

1. The Reactor Coolant System (RCS) recirculation flow control rod block is neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The RCS recirculation flow control rod block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The RCS recirculation flow control rod block signal is not a part of a primary success path in the mitigation of a DBA or transient.

As discussed in section 3.5 and summarized in Table 4-1 (item 140) of NEDO-31466, the loss of the RCS recirculation flow control rod block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the control rod block LCO and Surveillances applicable to RCS recirculation flow instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

LCO Statement:

The radiation monitoring instrumentation channels shown in Table 3.3.7.1-1 shall be OPERABLE, with their alarm/trip setpoints within the specified limits.

3/4.3.7.1.2 Area Monitors

Discussion:

The area radiation monitors are used to indicate when the radiation in the new fuel storage vault, spent fuel storage pool, or main control room areas has exceeded its allowable setpoint. There are no automatic functions that are performed by these instruments. The instruments are not used to mitigate a design basis accident (DBA) or transient. Information provided by these instruments on the radiation levels would have limited or no use in identifying/assessing core damage.

Comparison to Screening Criteria:

1. These monitors are neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The monitored parameters are not assumed as initial conditions of a DBA or transient analysis that assumes the failure of, or presents a challenge to the integrity of a fission product barrier.

3. These monitors do not act as part of a primary success path in the mitigation of a DBA or transient that assumes the failure of, or presents a challenge to the integrity of a fission product barrier.

As discussed in Section 3.5 and summarized in Table 4-1 (item 150) of NEDO-31466, the loss of these monitors was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

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

3/4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

LCO Statement:

The seismic monitoring instrumentation shown in Table 3.3.7.2-1 shall be OPERABLE.

Discussion:

In the event of an earthquake, seismic instrumentation is required to permit comparison of the measured response to that used in the design basis of the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. There is no automatic action that these instruments perform during a seismic event. Since this is determined after the event has occurred, it has no bearing on the mitigation of any design basis accident (DBA). The magnitude of the earthquake can also be obtained from the National Earthquake Information Service or other sources.

Comparison to Screening Criteria:

1. These instruments are neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

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

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

As discussed in Section 3.5 and summarized in Table 4-1 (item 151) of NEDO-31466, the loss of seismic monitoring instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

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

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

LCO Statement:

The meteorological monitoring instrumentation channels shown in Table 3.3.7.3-1 shall be OPERABLE.

Discussion:

Meteorological instrumentation is used to measure environmental parameters which may affect distribution of fission products and gases following a design basis accident (DBA), but it is not an input assumption for any DBA analysis and does not mitigate the accident. There is no automatic action that these instruments perform during any event. Meteorological information is required to evaluate the need for initiating protective measures to protect the health and safety of the public in the event of an accident. However, this information can be obtained from the National Weather Service or other sources.

Comparison to Screening Criteria:

1. These instruments are neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. These monitored parameters—wind direction, speed, air temperature, and air temperature differences—are not process variables that are initial conditions in a DBA or transient analysis.

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

As discussed in Section 3.5 and summarized in Table 4-1 (item 152) of NEDO-31466, the loss of meteorological monitoring instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

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

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

LCO Statement:

The accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

Discussion:

Each individual accident monitoring parameter has a specific purpose, however, the general purpose for all accident monitoring instrumentation is to provide sufficient information to assess plant response in the event of an accident; i.e., automatic safety systems are performing properly and deviations from expected accident course are minimal.

Comparison to Screening Criteria:

The NRC position on application of the screening criteria to post-accident monitoring instrumentation is documented in letter dated May 7, 1988, from T.E. Murley (NRC) to R.F. Janecek (BWROG). 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 SER on Regulatory Guide 1.97, and all Regulatory Guide 1.97 Category 1 instruments. Accordingly, this position has been applied to the CPS Regulatory Guide 1.97 instruments. Those instruments meeting this criteria have remained in Technical Specifications, and those instruments not meeting the criteria have been relocated from the Technical Specifications to plant controlled documents.

The following summarizes the application of the NRC position to CPS.

From SER Supplement 5, dated January, 1986, Subject: SER Related to the Operation of Clinton Power Station Unit 1 and Regulatory Guide 1.97:

Type A Variables

1. Reactor Vessel Pressure
2. Reactor Vessel Water Level
3. Suppression Pool Water Level
4. Suppression Pool Water Temperature
5. Drywell Pressure
6. Drywell/Containment Hydrogen and Oxygen Concentration Analyzer and Monitor

Other Type, Category 1 Variables

1. Containment Pressure
2. Containment/Drywell High Range Gross Gamma Radiation Monitors
3. Primary Containment Isolation Valve Position Indication

Conclusion

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

1. Drywell Air Temperature
2. Containment Temperature
3. Safety/Relief Valve Acoustic Monitor
4. HVAC Stack High Range Radioactivity Monitor
5. SGTS Exhaust High Range Radioactivity Monitor

3/4.3.7.7 TRAVERSING IN-CORE PROBE

LCO Statement:

The traversing in-core probe system shall be OPERABLE, with:

Discussion:

The traversing in-core probe (TIP) system is used only for calibration of the LPRM detectors. The TIP System is positioned axially and radially throughout the core to calibrate the local power range monitors (LPRMs). When not in use, the TIP instruments are retracted into a storage position inside the drywell wall penetrations. The TIP System supports the OPERABILITY of the LPRMs. With LPRM OPERABILITY addressed, there is no need to address the TIP System in the Technical Specifications.

Comparison to Screening Criteria:

1. The TIP System is neither 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 TIP System is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

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

As discussed in Section 3.5 and summarized in Table 4-1 (item 183) of NEDO-31466, the loss of the TIP System was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

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

3/4.3.7.8 CHLORINE DETECTION SYSTEM

LCO Statement:

Two independent chlorine detection channels shall be OPERABLE, with their trip setpoints adjusted to actuate at a chlorine concentration of $\leq$ 5 ppm.

Discussion:

The Chlorine Detection System is used to isolate the control room upon detection of a high concentration of chlorine. The chlorine release would not be a result of a design basis accident (DBA) or transient; thus, the instruments do not perform any required function during a design basis event. Amendment No. 12 to the CPS Technical Specifications incorporated provisions that this LCO would no longer be applicable after all chlorine containers having a capacity of 100 pounds or greater are removed from the site. Because this condition has been met, this LCO is no longer required to be met at CPS.

Comparison to Screening Criteria:

1. The Chlorine Detection System is neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The Chlorine Detection System is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

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

As discussed in Section 3.5 and summarized in Table 4-1 (item 184) of NEDO-31466, the loss of the Chlorine Detection System was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

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

3/4.3.7.10 LOOSE-PART DETECTION SYSTEM

LCO Statement:

The loose-part detection system shall be OPERABLE.

Discussion:

The Loose-Part Detection System is used to detect loose parts in the reactor vessel. The instrumentation does not indicate that there is a degradation in the primary pressure boundary but indicates that there might be a remote chance of damage to a component due to a loose part. The potential of fuel failure due to fuel bundle flow blockage from a lost part will be detected by the radiation monitors in the offgas stream.

Comparison to Screening Criteria:

1. The Loose-Part Detection System is neither 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 Loose-Part Detection System is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The Loose-Part Detection System is not part of a primary success path in the mitigation of a DBA or transient.

As discussed in Section 3.5 and summarized in Table 4-1 (item 187) of NEDO-31466, the loss of the Loose-Part Detection System was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

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

3/4.3.7.11 MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING INSTRUMENTATION

LCO Statement:

At least one main condenser offgas treatment system explosive gas monitoring instrumentation channel shall be OPERABLE, with its alarm/trip setpoint set to ensure that the limits of Specification 3.7.8.1 are not exceeded.

Discussion:

The explosive gas monitor Specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the main condenser offgas treatment system is adequately monitored, which will help ensure that the concentration is maintained below the flammability limit of hydrogen. However, the offgas system is designed to contain detonations and will not affect the function of any safety related equipment. The concentration of hydrogen in the offgas stream is not an initial assumption of any design basis accident (DBA) or transient analysis.

Comparison to Screening Criteria:

1. The explosive gas mixture indication is neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The explosive gas mixture is not a process variable that is an initial condition of a DBA or transient analyses.

3. The explosive gas mixture indication is not utilized in any capacity in a primary success path in the mitigation of a DBA or transient.

As discussed in Section 3.5 and summarized in Table 4-1 (item 306) of NEDO-31466, an explosive gas mixture in the Offgas Treatment System was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with this assessment.

Conclusion:

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

3/4.3.9.2 FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM

LCO Statement:

The Feedwater System/Main Turbine Trip System shall be OPERABLE.

Discussion:

The Feedwater System/Main Turbine Trip on Reactor Vessel Water Level—High, Level 8 is used in the Design Basis transient analysis for plants that do not have a scram from Reactor Protection System Reactor Vessel Level 8. Clinton (like all other BWR-6 plants) has a direct scram on Reactor Vessel Water Level 8. The Design Basis transient analysis does not require the scram that would be received from the trip of the main turbine. Consequently, this LCO does not serve any primary safety function (i.e., detection or mitigation of a design basis accident (DBA) or transient).

Comparison to Screening Criteria:

1. The Feedwater System/Main Turbine Trip System is neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The Feedwater System/Main Turbine Trip System is not capable of monitoring a process variable that is an initial condition of a DBA or transient analyses.

3. The Feedwater System/Main Turbine Trip System is not part of a primary success path in the mitigation of a DBA or transient.

As discussed in Section 3.5 and summarized in Table 4-1 (item 194) of NEDO-31466, the loss of the Feedwater System/Main Turbine Trip System was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Feedwater System/Main Turbine Trip System LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.9.3 SUPPRESSION POOL MAKEUP SYSTEM (SPMS)

LCO Statement:

The Suppression Pool Makeup System (SPMS) Mode Switch Permissive shall be OPERABLE.

Discussion:

The SPMS Mode Switch Permissive Function is an operational function only and is not considered in any design basis accident (DBA) or transient analysis. In addition, this switch permissive function is controlled under administrative controls to assure the appropriate position of the switch is maintained.

Comparison to Screening Criteria:

1. The SPMS Mode Switch Permissive is neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The SPMS Mode Switch Permissive is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The SPMS Mode Switch Permissive is not part of a primary success path in the mitigation of a DBA or transient.

Conclusion:

Since the screening criteria have not been satisfied, the SPMS Mode Switch Permissive Function may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.3.10 NUCLEAR SYSTEM PROTECTION SYSTEM—SELF TEST SYSTEM

LCO Statement:

The Self Test SYSTEM (STS) of the Nuclear System Protection System shall be OPERABLE and operating in the fully automatic mode.

Discussion:

The primary purpose of the Self Test System is to enhance the availability of the Nuclear System Protection System by optimizing the time to detect and determine the location of a failure in the functional system. The Self Test System is used for post-maintenance testing and to augment conventional testing methods to perform various surveillance testing functions.

Comparison to Screening Criteria:

1. The Nuclear System Protection System—Self Test System is neither 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 Nuclear System Protection System—Self Test System is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The Nuclear System Protection System—Self Test System is not part of a primary success path in the mitigation of a DBA or transient.

As discussed in Section 6 and summarized in Table 4-1 (item 312) of NEDO-31466, Supplement 1, the loss of the Nuclear System Protection System—Self Test System was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Nuclear System Protection System—Self Test System LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.4.4 CHEMISTRY

LCO Statement:

The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.4-1.

Discussion:

Poor coolant water chemistry contributes to the long-term degradation of system materials of construction and thus is not of immediate importance to the plant 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. Severe chemistry transients have resulted in failure of thin walled LPRM instrument dry tubes in a relatively short period of time. However,these LPRM dry tube failures result in loss of the LPRM function and are readily detectable. In summary, the chemistry monitoring activity is of a long-term preventative purpose rather than mitigative.

Comparison to Screening Criteria:

1. Reactor coolant water chemistry is neither 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 used to monitor a process variable that is an initial condition of a DBA or transient analyses.

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

As discussed in Section 3.5 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. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

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

3/4.4.8 STRUCTURAL INTEGRITY

LCO Statement:

The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.8.

Discussion:

The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained throughout the component's life. Other Technical Specifications require important systems to be OPERABLE (for example, ECCS 3/4.5.1) and in a ready state for mitigative action. This Technical Specification is more directed toward prevention of component degradation and continued long-term maintenance of acceptable structural conditions. Hence, it is not necessary to retain this Specification to ensure immediate OPERABILITY of safety systems.

Comparison to Screening Criteria:

1. The inspections stipulated by this Specification are neither 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 inspections stipulated by this Specification do not monitor process variables that are initial assumptions in a DBA or transient analysis.

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

As discussed in Section 3.5 and summarized in Table 4-1 (item 216) of NEDO-31466, the assurance of OPERABILITY of the entire system as verified in the system OPERABILITY Specification dominates the risk contribution of the system. The lack of a long-term assurance of structural integrity as stipulated by this Specification was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

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

3/4.7.4 SNUBBERS

LCO Statement:

All snubbers shall be OPERABLE.

Discussion:

Snubbers are included in the plant design to ensure that the structural integrity of the reactor coolant system and other safety related systems are maintained during and after a seismic or other dynamic loading event. The snubbers are considered a part of the piping system. They serve as an aid to prevent piping failure, but do not mitigate piping failure should it occur. Also, the failure of a snubber on a particular pipe cannot, by itself, cause the pipe to fail. Consequently, the snubbers do not meet any of the criteria, since they are not utilized as part of the primary success path in detecting or mitigating the consequences of a design basis accident (DBA) or transient event. Additionally, the surveillance and maintenance of the snubbers can be controlled by sources other than the plant Technical Specifications.

Comparison to Screening Criteria:

1. Snubbers are neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. Snubbers are not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

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

As discussed in Section 3.5 and summarized in Table 4-1 (item 266) of NEDO-31466, the loss of snubbers was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

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

3/4.7.5 SEALED SOURCE CONTAMINATION

LCO Statement:

Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 10 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

Discussion:

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

Comparison to Screening Criteria:

1. Sealed source contamination is neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Sealed source contamination is not a process variable that is an initial condition of a DBA or transient analyses.

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

As discussed in Section 3.5 and summarized in Table 4-1 (item 267) of NEDO-31466, the sealed source contamination being not within limits was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

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

3/4.8.4.1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LCO Statement:

Primary and backup containment penetration conductor overcurrent protective devices associated with each primary containment electrical penetration circuit shall be OPERABLE. The scope of these protective devices excludes those circuits for which credible fault currents would not exceed the electrical penetrations' design ratings.

Discussion:

The primary feature of these protective devices is to open the control and/or power circuit whenever the load conditions exceed the preset current demands. This is to protect the circuit conductors against damage or failure due to overcurrent heating effects.

The continuous monitoring of the operating status of the overcurrent protective devices is impracticable and not covered as part of the control room monitoring, except after trip condition indication.

In the event of failure of this protective device to trip the circuit, the upstream protective device is expected to operate and isolate the faulty circuit. Thus, the upper level (back-up) protection will protect the circuit conductors. In the worst-case fault condition, a single division of protective functions can be lost. However, this scenario is covered under a single failure criterion.

The overcurrent protective devices ensure the pressure integrity of the containment penetration. With failure of the device it is postulated that the wire insulation will degrade, resulting in a containment leak path during a LOCA. However, the protection provided by these devices is not a process variable and is not considered as part of the primary success path. Containment penetration degradation will be identified during the normal containment leak rate tests required by 10 CFR Part 50, Appendix J.

Comparison to Screening Criteria:

1. The containment penetration conductor overcurrent protective devices are neither 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 containment penetration conductor overcurrent protective devices do not monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The specific circuits of the containment penetration conductor overcurrent protective devices are not part of a primary success path in the mitigation of a DBA or transient.

As discussed in Section 3.5 and summarized in Table 4-1 (item 276) of NEDO-31466, loss of the containment penetration conductor overcurrent protective devices was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Containment Penetration Conductor Overcurrent Protective Devices LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.8.4.2 MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

LCO Statement:

The thermal overload protection of each valve in safety systems with a bypass device(s) integral with the motor starter shall be bypassed continuously for those directions for which the valve performs an active safety function.

Discussion:

For valves with thermal overload protection (i.e., trip on overload condition), the valve function should be accomplished prior to overload trip. The overload protection for these valves is meant to take precedence over the valve function. If an overload condition occurs during valve operation, the electrical circuit will open to protect the equipment. In case of failure of overload protection operation to disconnect the load, the equipment may suffer potential damage. This may impact the OPERABILITY of the system containing the valve. Accordingly, the system LCO would address the overall system OPERABILITY, and not the OPERABILITY of a support system. Additionally, the surveillance and maintenance of the motor operated valves thermal overload protection can be controlled by sources other than the plant Technical Specifications.

Comparison to Screening Criteria:

1. Motor operated valve thermal overload protection is neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Motor operated valve thermal overload protection does not monitor a process variable that is an initial condition of a DBA or transient analyses.

3. Actuation of a motor operated valve's thermal overload protection is not part of a primary success path in the mitigation of a DBA or transient. The supported system (e.g., ECCS) may be part of a success path and is then retained in the Technical Specifications. However, motor operated valve thermal overload protection retention in the Technical Specifications is not necessary as its function is confirmed in the OPERABILITY of the supported system.

Probabilistic Risk Assessments (PRAs) address system risk contribution and identify systems which can be significant risk contributors to core damage and offsite releases. Subcomponent risk contribution from motor operated valve thermal overload protection malfunction is not addressed as the inoperability of the supported system and subsequent risk contribution is bounding.

Conclusion:

Since the screening criteria have not been satisfied, the Motor Operated Valve Thermal Overload Protection LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.9.5 COMMUNICATIONS

LCO Statement:

Direct communication shall be maintained between the control room and refueling platform personnel.

Discussion:

Communication between the control room and refueling floor personnel is maintained to ensure that refueling personnel can be promptly informed of significant changes in the plant status or core reactivity condition during refueling. The communications allow for coordination of activities that require interaction between the control room and refueling floor personnel (such as the insertion of a control rod prior to loading fuel). However, the refueling system design accident or transient response does not take credit for communications and is designed to ensure safe refueling operations.

Comparison to Screening Criteria:

1. Communications during any mode of plant operation is neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Communications during any mode of plant operation is not used to indicate status of, or monitor a process variable that is an initial condition of a DBA or transient analyses.

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

As discussed in Section 3.5 and summarized in Table 4-1 (item 286) of NEDO-31466, the loss of communication was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

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

3/4.9.6 FUEL HANDLING EQUIPMENT

LCO Statement:

3.9.6.1 The refueling platform shall be OPERABLE and used for handling fuel assemblies or control rods within the reactor pressure vessel.

3.9.6.2 The auxiliary platform shall be OPERABLE.

Discussion:

OPERABILITY of the refueling equipment (refueling and auxiliary platforms) ensures that only the proper hoists of the refueling and auxiliary platforms will be used to handle fuel within the reactor pressure vessel or fuel pool, hoists have sufficient load capacity for handling fuel assemblies and/or control rods and the core internals and pressure vessel are protected from excessive lifting force if they are inadvertently engaged during lifting operations. Although the interlocks designed to provide the above capabilities can prevent damage to the fuel handling equipment and core internals, they are not assumed to function to mitigate the consequences of a design basis accident (DBA).

Comparison to Screening Criteria:

1. The fuel handling equipment and associated instrumentation are neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The fuel handling equipment and associated instrumentation are not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

3. The fuel handling equipment and associated instrumentation are not part of a primary success path in the mitigation of a DBA or transient.

As discussed in Section 3.5 and summarized in Table 4-1 (item 287) of NEDO-31466, the refueling equipment and associated instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Fuel Handling Equipment LCOs and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.9.7 CRANE TRAVEL—SPENT FUEL STORAGE POOL, UPPER CONTAINMENT FUEL POOL, AND NEW FUEL STORAGE VAULT

LCO Statement:

Loads in excess of 1000 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks, upper containment fuel pool racks or new fuel storage vault racks.

Discussion:

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the storage pools ensures that in the event the load is dropped, the activity release will be limited to that contained in a single fuel assembly and any possible distortion of the fuel in the storage racks will not result in a critical array. Administrative monitoring of loads moving over the fuel storage racks serves as a backup to the crane interlocks.

Although this LCO supports the maximum refueling accident assumption in the DBA, these types of limitations are adequately controlled by administrative controls. Therefore, the criteria for Technical Specification retention are not satisfied.

Comparison to Screening Criteria:

1. The crane travel load limits are neither 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 maximum severity assumed for the fuel handling DBA is limited by the load limits placed on the crane travel. These crane travel limits are not, however, process variables monitored and controlled by the operator. They may be interlocks and/or physical stops and are addressed by administrative controls. Criterion 2 is thus not satisfied.

3. The crane travel load 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.

Traditional Probabilistic Risk Assessments (PRAs) do not review risks associated with the spent fuel storage pool.

Conclusion:

Since the screening criteria have not been satisfied, the Crane Travel—Spent Fuel Storage Pool, Upper Containment Fuel Pool, and New Fuel Storage Vault LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.9.12 INCLINED FUEL TRANSFER SYSTEM

LCO Statement:

The inclined fuel transfer system (IFTS) may be in operation provided that:

a. The access doors of all rooms through which the transfer system penetrates are closed and locked.

b. All access door interlocks are OPERABLE.

c. The blocking valve located in the fuel building IFTS hydraulic power unit is OPERABLE.

d. At least one IFTS carriage position indicator is OPERABLE at each carriage position and at least one liquid level sensor is OPERABLE.

e. Any keylock switch that provides IFTS access control-transfer system lockout is OPERABLE.

Discussion:

The IFTS transfers fuel from the secondary containment (fuel building) into primary containment (upper fuel pool). The purpose of the IFTS Specification is to limit personnel access to potentially high-radiation areas of the system. This requirement is not an assumption of any design basis accident (DBA), but helps to ensure that 10 CFR 20 limits are not exceeded.

Comparison to Screening Criteria:

1. The IFTS is neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The IFTS is not used to monitor a process variable that is an initial condition of a DBA or transient analyses.

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

As discussed in Section 3.5 and summarized in Table 4-1 (item 294) of NEDO-31466, the IFTS was found to be a non-significant risk contributor to core damage frequency and offsite releases. IP has reviewed this evaluation, considers it applicable to CPS, and concurs with the assessment.

Conclusion:

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