H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
APPLICATION OF SELECTION CRITERIA

APPLICATION OF SELECTION CRITERIA TO THE

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2

TECHNICAL SPECIFICATIONS

1. INTRODUCTION

The purpose of this document is to confirm the results of the Westinghouse Owners Group application of the Technical Specification selection criteria on a plant specific basis for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. Carolina Power & Light (CP&L) Company has reviewed the application of the selection criteria to each of the Technical Specifications utilized in report WCAP-11618, "Methodically Engineered, Restructured and Improved, Technical Specifications, Merits Program - Phase II Task 5, Criteria Application" (Reference 1) including Addendum 1, NRC Staff Review of NSSS Vendor Owners Group Application of the Commission's Interim Policy Statement to Standard Technical Specifications, Newton/Murley letter dated May 9, 1988 and as revised in NUREG-1431, Revision 1 "Standard Technical Specifications, Westinghouse Plants," (Reference 2) and applied the criteria to each of the current HBRSEP, Unit No. 2 Technical Specifications. Additionally, in accordance with 10 CFR 50.36 (c)(2)(ii) and the NRC final policy statement (Reference 3), this confirmation of the application of selection criteria includes confirming the risk insights from Probabilistic Risk Assessment (PRA) evaluations, provided in the WCAP document, as applicable to HBRSEP, Unit No. 2.

2. SELECTION CRITERIA

CP&L used the selection criteria provided in the NRC Final Policy Statement on Technical Specification Improvements of July 22, 1993 (Reference 3) to develop the results contained in the attached matrix. Probabilistic Risk Assessment (PRA) insights as used in WCAP-11618 were used, confirmed by CP&L, and are discussed in the next section of this report. The selection criteria and discussion provided Reference 3 are as follows:

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

Discussion of Criterion 1: A basic concept in the adequate protection of the public health and safety is the prevention of accidents. Instrumentation is installed to detect significant abnormal degradation of the reactor coolant pressure boundary so as to allow operator actions to either correct the condition or to shut down the plant safely, thus reducing the likelihood of a loss-of-coolant accident. This criterion is intended to ensure that Technical Specifications control those instruments specifically installed to detect excessive reactor coolant system leakage. This criterion should not, however, be interpreted to include instrumentation to detect precursors to reactor coolant pressure boundary leakage or instrumentation to identify the source of actual leakage (e.g., loose parts monitor, seismic instrumentation, valve position indicators).

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 2: Another basic concept in the adequate protection of the public health and safety is that the plant shall be operated within the bounds of the initial conditions assumed in the existing Design Basis Accident and Transient analyses and that the plant will be operated to preclude unanalyzed transients and accidents. Analyses consist of postulated events, analyzed in the FSAR, for which a structure, system, or component must meet specified functional goals. These analyses are contained in Chapters 6 and 15 of the FSAR (or equivalent chapters) and are identified as Condition II, III, or IV events (ANSI N18.2) (or equivalent) that either assume the failure of or present a challenge to the integrity of a fission product barrier.

As used in Criterion 2, process variables are only those parameters for which specific values or ranges of values have been chosen as reference bounds in the Design Basis Accident or Transient analyses and which are monitored and controlled during power operation such that process values remain within the analysis bounds. Process variables captured by Criterion 2 are not, however, limited to only those directly monitored and controlled from the control room. These could also include other features or characteristics that are specifically assumed in Design Basis Accident or Transient analyses if they cannot be directly observed in the control room (e.g., moderator temperature coefficient and hot channel factors).

The purpose of this criterion is to capture those process variables that have initial values assumed in the Design Basis Accident and Transient analyses, and which are monitored and controlled during power operation. As long as these variables are maintained within the established values, risk to the public safety is presumed to be acceptably low. This criterion also includes active design features (e.g., high pressure/low pressure system valves and interlocks) needed to preclude unanalyzed accidents and transients.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 3: A third concept in the adequate protection of the public health and safety is that in the event that a postulated Design Basis Accident or Transient should occur, structures, systems, and components are available to function or to actuate in order to mitigate the consequences of the Design Basis Accident or Transient. Safety sequence analyses or their equivalent have been performed in recent years and provide a method of presenting the plant response to an accident. These can be used to define the primary success paths.

A safety sequence analysis is a systematic examination of the actions required to mitigate the consequences of events considered in the plant's Design Basis Accident and Transient analyses, as presented in Chapters 6 and 15 of the plant's FSAR (or equivalent chapters). Such a safety sequence analysis considers all applicable events, whether explicitly or implicitly presented. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria), so that the plant response to Design Basis Accidents and Transients limits the consequences of these events to within the appropriate acceptance criteria.

It is the intent of this criterion to capture into Technical Specifications only those structures, systems, and components that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path for a particular mode of operation does not include backup and diverse equipment (e.g., rod withdrawal block which is a backup to the average power range monitor high flux trip in the startup mode, safety valves which are backup to low temperature overpressure relief valves during cold shutdown).

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

Discussion of Criterion 4: It is the Commission's policy that licensees retain in their Technical Specifications LCOs, action statements, and Surveillance Requirements for the following systems (as applicable), which operating experience and PSA have generally shown

to be significant to public health and safety and any other structures, systems, or components that meet this criterion:

- Reactor Core Isolation Cooling/Isolation Condenser,
- Residual Heat Removal,
- Standby Liquid Control, and
- Recirculation Pump Trip.

The Commission recognizes that other structures, systems, or components may meet this criterion. Plant- and design-specific PSA's have yielded valuable insight to unique plant vulnerabilities not fully recognized in the safety analysis report Design Basis Accident or Transient analyses. It is the intent of this criterion that those requirements that PSA or operating experience exposes as significant to public health and safety, consistent with the Commission's Safety Goal and Severe Accident Policies, be retained or included in the Technical Specifications.

The Commission expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA or risk survey and any available literature on risk insights and PSAs. This material should be employed to strengthen the technical bases for those requirements that remain in Technical Specifications, when applicable, and to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk. Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical Specifications related submittals. Further, as part of the Commission's ongoing program of improving technical specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements."

3. PROBABILISTIC RISK ASSESSMENT INSIGHTS

Introduction and Objectives

Reference 3 includes an NRC expectation that CP&L 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 (TS) proposed for relocation to other plant controlled documents will be maintained under the 10 CFR 50.59, safety evaluation review program. Relocated specifications have been compared to a variety of Probabilistic Risk Assessment (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. Those TS proposed to remain part of the Improved Technical Specifications (ITS) were not reviewed. This review was accomplished in WCAP-11618 (Reference 1), except where discussed in Appendix A, Justification for Specification Relation, and has been confirmed by CP&L for those specifications to be relocated.

Assumptions and Approach

The WCAP-11618 evaluation of the risk impact of the TS that are relocation candidates was based on the following:

a. It was assumed that any of the TS that were to be relocated would be transferred to other documents subject to control by the utility under the 10 CFR 50.59 process.

b. The risk criteria used in determining the disposition of a TS were the following:

   1. If the TS contained constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk, it should be retained.

   2. If the TS included items involved in one of these dominant sequences but had an insignificant impact on the probability or severity of that sequence, it was proposed to be relocated to another controlled document.

   3. If the TS was not involved in risk dominant sequences, it was proposed to be relocated to another controlled document.

c. The measures related to risk used in this evaluation were core melt frequency and

off-site health effects. These measures were consistent with the NRC Final Policy Statement on TS and the Safety Goal and Severe Accident Policy Statements.

d. The criteria used to determine if a sequence was risk dominant was the following:

For core melt, any sequence whose frequency was commonly found to be greater than 1 X 10^{-6} per reactor year was maintained as a possible dominant sequence as a conservative first cut. This was roughly 2% of the total core melt frequency of 5 X 10^{-5} for typical PRAs. Each specific sequence identified in the screening of the TS was evaluated against the above conservative criterion to determine if it was risk dominant.

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

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

Systems identified in Tables 3 and 4 of Section 4.0 of WCAP-11618 that contributed significantly to risk as defined in Paragraph d above were listed in Tables 3A, 3B, 4A and 4B of Section 4.0. These identified systems as well as sequences and the risk dominant initiating events from Tables 3 and 4 which were involved in typical dominant core melt and serious release sequences from formal risk assessments were used to screen the requirements of the TS reviewed. Those TS whose requirements were relevant to these systems, sequences, and initiating events were further evaluated for risk dominance. The remaining TS were evaluated on the basis of risk insights from references listed in Section 4.0, Appendix B of WCAP-11618. If the requirements of a TS were not found to be modeled in any reference and no significant issues were identified from a review of the risk insights, the conclusion was that it did not contain constraints of <u>prime importance</u> to limiting the likelihood or severity of sequences that are <u>commonly found</u> to <u>dominate</u> risk.

4. RESULTS OF APPLICATION OF SELECTION CRITERIA

The selection criteria from Section 2 were applied to the HBRSEP, Unit No. 2 TS. The attachment is a summary of that application indicating which Specifications are being retained or relocated. Discussions that document the rationale for the relocation of each Specification which failed to meet the selection criteria are provided in Appendix A. No Significant Hazards Considerations (10 CFR 92) evaluations for those Specifications relocated are provided with the Discussion of Changes for the specific TS. CP&L has relocated those Specifications identified as not satisfying the criteria to licensee controlled documents whose changes are governed by 10 CFR 50.59.

5. REFERENCES

1. WCAP-11618 (and Addendum 1), "Methodically Engineered Restructured and Improved Technical Specifications, MERITS Program—Phase II Task 5, Criteria Application," November 1987.

2. NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 1, April 1995.

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

SUMMARY DISPOSITION MATRIX

CURRENT TS NUMBER	DESCRIPTION	NUREG-0452 Rev. 4	ITS NUMBER	RETAINED-CRITERIA FOR INCLUSION	NOTES
1.0	DEFINITIONS	1.0	1.1	Yes	Definitions for selected terms used in the Current TS (CTS) are provided to improve understanding and ensure consistent application. Application of the TS selection criteria to these definitions is not appropriate. However, definitions for those terms that remain in the TS following the application of the selection criteria are retained.
	SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS				
2.1	SAFETY LIMIT, REACTOR CORE	2.1.1 3.4.1.1	2.1.1 3.4.4	Yes	Application of TS selection criteria to Safety Limits is not appropriate. The safety limits for the reactor core are retained in the ITS.
2.2	SAFETY LIMIT, REACTOR COOLANT SYSTEM PRESSURE	2.1.2	2.1.2	Yes	Application of TS selection criteria to Safety Limits is not appropriate. The safety limits for the Reactor Coolant System pressure are retained in the ITS.
2.3	LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION	2.2	3.3.1	Yes-3	The limiting safety settings for protective instrumentation function to actuate the Reactor Protection System (RPS) to mitigate the consequences of Design Basis Accidents (DBAs) and/or transients.
3.0	LIMITING CONDITIONS FOR OPERATION	3.0.3	3.0.3	Yes	This Specification provides guidance applicable to one or more Limiting Conditions for Operation (LCOs). Direct application of the TS selection criteria is not appropriate. The general requirements of CTS Section 3.0 are retained in the ITS.
	REACTOR COOLANT SYSTEM				
3.1.1.1 3.1.1.2 3.1.1.3	COOLANT PUMPS STEAM GENERATOR PRESSURIZER (PZR)	3.4.1.1 3.4.1.2 3.4.1.3	3.4.4 3.4.5 3.4.6 3.4.7 3.4.9 3.4.10	Yes-2, 3,	Operation of the reactor coolant pumps during various plant modes is an initial assumption in accident analyses. The number of operable steam generators is an initial assumption in the accident analyses.
3.1.1.4	REACTOR COOLANT SYSTEM VENT PATH	3.4.11	Relocated	No	See Appendix A page A-1.
3.1.1.5	RELIEF VALVES	3.4.4	3.4.11	Yes	The PORV is used to reduce primary system pressure and was incorporated into the TS in response to GL 90-06.
	HEATUP AND COOLDOWN				
3.1.2.1 3.1.2.4	REACTOR COOLANT SYSTEM	3.4.9.1 3.4.9.3	3.4.3 3.4.12	Yes-2	Establishes initial conditions such that operation is prohibited in areas or at temperature change rates that might cause undetected flaws to propagate in turn challenging the reactor coolant pressure boundary integrity.

CURRENT TS NUMBER	DESCRIPTION	NUREG-0452 Rev. 4	ITS NUMBER	RETAINED-CRITERIA FOR INCLUSION	NOTES
3.1.2.2	STEAM GENERATOR	3.7.2	Relocated	No	See Appendix A page A-3.
3.1.2.3	PRESSURIZER	3.4.9.2	Relocated	No	See Appendix A page A-5.
3.1.3	MINIMUM CONDITIONS FOR CRITICALITY	3.1.1.3 3.1.1.4 3.4.3	3.1.3 3.4.2 3.4.9	Yes-2	Establishes operating restrictions such that operation is bounded by the accident analysis.
3.1.4	MAXIMUM REACTOR COOLANT ACTIVITY	3.4.8	3.4.16	Yes-2	Establishes operating restrictions such that operation is bounded by the accident analysis.
3.1.5	LEAKAGE	3.4.6.2	3.4.13 3.4.14	Yes-2	Establishes operating restrictions such that operation is bounded by the accident analysis.
3.1.6 and Table 4.1-2, Item 1 (Cl and O_2 limits)	MAXIMUM REACTOR COOLANT OXYGEN AND CHLORIDE CONCENTRATION	3.4.7	Relocated	No	See Appendix A page A-6.
3.2	CHEMICAL AND VOLUME CONTROL SYSTEM	3.5.2		N/A	This specification is being relocated by separate application
	EMERGENCY CORE COOLING SYSTEM, AUXILIARY COOLING SYSTEMS, AIR RECIRCULATION FAN COOLERS, CONTAINMENT SPRAY POST ACCIDENT CONTAINMENT VENTING SYSTEM AND ISOLATION SEAL WATER SYSTEM				
3.3.1.1 3.3.1.2 3.3.1.3	SAFETY INJECTION AND RESIDUAL HEAT REMOVAL SYSTEMS	3.5.5 3.5.1 3.5.2 3.4.1.1 3.5.3 3.4.9.3	3.5.4 3.5.1 3.5.2 3.4.4 3.5.3 3.4.12	Yes-3 & 2	The ECCS systems function to provide cooling water to the reactor core to mitigate design basis accidents and transients. The limitation regarding safety injection pump breaker positions with temperature < 350 °F is necessary to establish conditions such that operation is bounded by the accident analysis (LTOP concern).
3.3.1.4	RHR LOOPS - COLD SHUTDOWN	3.5.3	3.4.7 3.4.8	Yes-4	RHR cooling in cold shutdown is identified in the NRC policy statement as an important contributor to risk reduction

CURRENT TS NUMBER	DESCRIPTION	NUREG-0452 Rev. 4	ITS NUMBER	RETAINED-CRITERIA FOR INCLUSION	NOTES
3.3.2.1	CONTAINMENT COOLING AND IODINE REMOVAL SYSTEMS	3.6.2 3.6.3	3.6.7 3.6.6	Yes-3	Containment Cooling functions to limit the containment post accident pressure to within design limits as well as reduce containment leakage rates. The Iodine removal system functions to limit the release of radioactive material to the environs.
3.3.3	COMPONENT COOLING SYSTEM	3.7.3	3.7.6	Yes-3	The Component Cooling System (CCS) functions to remove post LOCA heat loads from the containment sump during the recirculation phase. The CCS also functions to cool the unit from RHR entry conditions to cold shutdown during normal and post accident conditions.
3.3.4	SERVICE WATER SYSTEM	3.7.4	3.7.7	Yes-3	The Service Water System (SWS) functions in conjunction with the CCS to remove post LOCA heat loads from the containment sump during the recirculation phase. The SWS in conjunction with CCS also functions to cool the unit from RHR entry conditions to cold shutdown during normal and post accident conditions.
3.3.5	POST ACCIDENT CONTAINMENT VENTING SYSTEM	3.6.5	Relocated	No	See Appendix A page A-8.
3.3.6	ISOLATION SEAL WATER SYSTEM	3.6.1.4	3.6.8	Yes-3	Isolation valve seal water functions to assure effectiveness of selected isolation valves during conditions which require containment penetrations by providing a seal water at the valves.
3.3.7	EXTENDED MAINTENANCE	N/A	Deleted	No	Deleted, see Extended Maintenance technical change discussion in the Discussion of Changes for CTS 3.3.7.
	SECONDARY STEAM AND POWER CONVERSION SYSTEM				
3.4.1 3.4.3 3.4.4 3.4.5 3.4.6	SECONDARY STEAM AND POWER CONVERSION SYSTEM	3.7.1 3.3.2	3.7.1 3.7.2 3.7.4 3.7.5 3.7.8 3.3.8	Yes-3	The Main Steam Relief Valves Function to limit secondary pressure during design basis events. The Auxiliary Feedwater System functions to remove decay heat during design basis events thus mitigating consequences of events which could result in over pressurization of the RCS pressure boundary. The Condensate Storage Tank or lake water function to provide cooling water to remove decay heat and cool down the unit for design basis events. The Main Steam Stop valves function to isolate steam flow from the secondary side of the steam generators following a high energy line break.
3.4.2	SECONDARY COOLANT SPECIFIC ACTIVITY	3.7.1	3.7.15	Yes-2	Secondary Coolant specific activity is limited to reduce the radiological consequences of a main steam line break.
	INSTRUMENTATION SYSTEMS				
3.5.1	OPERATIONAL SAFETY INSTRUMENTATION	3.3.1 3.3.2 3.3.3.6	3.3.1 3.3.2 3.3.3 3.3.6	Yes-3	The Reactor Trip System Instrumentation functions to maintain safety limits during operation and to mitigate the consequences of design basis events. The Engineered Safety Feature Actuation System functions to detect and initiate mitigation for design basis events. The Containment Purge and Exhaust Isolation Instrumentation function to ensure closure of the purge and exhaust valves to limit the radiological consequences of design basis accidents.

CURRENT TS NUMBER	DESCRIPTION	NUREG-0452 Rev. 4	ITS NUMBER	RETAINED-CRITERIA FOR INCLUSION	NOTES
3.5.1 Table 3.5-1 Table 3.5-3	ENGINEERED SAFETY FEATURE SYSTEM INSTRUMENTATION	3.3.2	3.3.2 3.3.5 3.3.6	Yes-3	The Engineered Safety Feature Actuation System functions to detect and initiate mitigation for design basis events.
3.5.1 Table 3.5-2	REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS	3.3.1	3.3.1	Yes-3 & 2	The Reactor Trip System Instrumentation functions to maintain safety limits during operation and to mitigate the consequences of design basis events.
3.5.1 Table 3.5-4	ISOLATION INSTRUMENTATION	3.3.2	3.3.2 3.3.6	Yes-3	The Isolation Instrumentation functions to provide isolation of containment atmosphere and process systems that penetrate containment from the environment to limit the release of radioactivity following design basis events.
3.5.1 Table 3.5-5, Table 4.1-1	INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT	3.3.3.6	3.3.3	Yes-3 & 4	RG 1.97 Type A and Category 1 Variables are retained. See Appendix A page A-13 for additional information.
3.5.2 Table 3.5-6 4.19.1	RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION	N/A	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-11. In accordance with Generic Letter 89-01, Programmatic Controls are included in ITS Administrative Controls Section.
3.5.3 Table 3.5-7 4.19.2	RADIOACTIVE GASEOUS EFFLUENT INSTRUMENTATION	N/A	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-13. Programmatic Controls are included in ITS Administrative Controls Section.
	CONTAINMENT SYSTEMS				
3.6.1	CONTAINMENT INTEGRITY	3.6.1 3.9.1	3.6.1 3.6.2 3.6.3 3.9.3	Yes-3, 2	The containment functions to limit radioactive material released to the environment following design basis accidents. Applicability of shutdown margin requirements function to require containment operability when reactor is critical.
3.6.2	INTERNAL PRESSURE	3.6.1.5	3.6.4	Yes-2	Containment pressure is an initial condition used in the analysis to establish maximum peak internal pressure following design basis events.
3.6.3	CONTAINMENT AUTOMATIC ISOLATION TRIP VALVES	3.6.4	3.6.3	Yes-3	Containment Isolation Valves function to support leak tightness of the containment to limit radioactive material released to the environment following design basis accidents.
3.6.4	CONTAINMENT PURGE AND VENT VALVES	3.6.1.10	3.6.3	Yes-3	Containment Isolation Valves function to support leak tightness of the containment to limit radioactive material released to the environment following design basis accidents.
3.7	AUXILIARY ELECTRIC SYSTEMS	3.8.1 3.8.2 3.8.3	3.8.1 3.8.3 3.8.4 3.8.9	Yes-3	The Auxiliary Electric Systems Function to ensure availability of power to equipment and systems used to mitigate the consequences of design basis events.
	REFUELING				

CURRENT TS NUMBER	DESCRIPTION	NUREG-0452 Rev. 4	ITS NUMBER	RETAINED-CRITERIA FOR INCLUSION	NOTES
3.8.1.a 3.8.1.b 3.8.1.d 3.8.1.e 3.8.1.f 3.8.1.j 3.8.1.k	REFUELING	3.9.8.1 3.3.3.1 3.9.2 3.9.8 3.9.1 3.9.12	3..3 3.9.5 3.9.2 3.9.4 3.9.5 3.9.1 3.7.11 3.9.1	Yes-2, 3 & 4	The Boron Concentration Limit Functions to ensure the reactivity condition of the core remains ≤ 0.95 % Δk/k during refueling operations. The source range nuclear instrumentation functions provide a signal to alert operators to unexpected changes in core reactivity. Containment closure along with purge and vent isolation functions to contain fission product release following fuel handling operations. The RHR system has been identified by the NRC as an important contributor to risk reduction.
3.8.1.c	CONTAINMENT RADIATION MONITORING DURING REFUELING	3.9.9	3.3.6	Yes-3	Containment Radiation Monitoring functions to isolate containment ventilation in the event of a fuel handling accident inside containment..
3.8.1.c	SPENT FUEL BUILDING RADIATION MONITORING DURING REFUELING	3.3.3.1	Relocated	No	See Appendix A page A-15.
3.8.1.g	COMMUNICATIONS DURING REFUELING	3.9.5	Relocated	No	See Appendix A page A-17.
3.8.1.h	TIME AFTER SHUTDOWN	3.9.3	Deleted	No	Deleted, see Decay Time technical change discussion in the Discussion of Changes for CTS 3.8.1.h.
3.8.1.i	CONTAINMENT PURGE ISOLATION, FILTRATION AND SPENT FUEL BUILDING FILTRATION	3.9.9 3.9.12	3.9.3 3.7.11	Yes-2, 3	Containment purge filtration functions to reduce the quantity of radioactive materials released after a fuel handling accident until the containment ventilation system is isolated. The isolation valves are used to isolate the containment in the event of a fuel handling accident inside containment. The spent fuel filtration system limits the offsite dose in the event of a fuel handling accident in the spent fuel pool.
3.8.1.i	CONTAINMENT PURGE FILTRATION	None	3.9.3	Yes-2	Containment purge filtration functions to reduce the quantity of radioactive materials released after a fuel handling accident until the containment ventilation system is isolated.
3.8.2	SPENT FUEL BUILDING FILTER SYSTEM	3.9.12	3.7.11	Yes-3	The Spent Fuel Building Filter System Functions to limit fission product radioactive release following design basis events.
3.8.3	SPENT FUEL POOL WATER TEMPERATURE	None	Relocated	No	See Appendix A page A-18.
3.8.4	SPENT FUEL CASK HANDLING CRANE	None	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-20.
	RADIOACTIVE EFFLUENTS				
3.9.1/4.10.1	COMPLIANCE WITH 10 CFR PART 20 - RADIOACTIVE MATERIALS IN LIQUID EFFLUENTS	None	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-22. In accordance with Generic Letter 89-01, Programmatic Controls are included in ITS Administrative Controls Section.

CURRENT TS NUMBER	DESCRIPTION	NUREG-0452 Rev. 4	ITS NUMBER	RETAINED-CRITERIA FOR INCLUSION	NOTES
3.9.2	COMPLIANCE WITH 10 CFR PART 50 - RADIOACTIVE MATERIALS IN LIQUID EFFLUENTS	None	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-24. In accordance with Generic Letter 89-01, Programmatic Controls are included in ITS Administrative Controls Section.
3.9.3/4.10.2	COMPLIANCE WITH 10 CFR PART 20 - RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS	None	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-26. In accordance with Generic Letter 89-01, Programmatic Controls are included in ITS Administrative Controls Section.
3.9.4/4.10.3	COMPLIANCE WITH 10 CFR PART 50 - RADIONOBLE GASES	None	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-28. In accordance with Generic Letter 89-01, Programmatic Controls are included in ITS Administrative Controls Section.
3.9.5/4.10.5	COMPLIANCE WITH 10 CFR PART 50 - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN RADIONOBLE GASES	None	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-30. In accordance with Generic Letter 89-01, Programmatic Controls are included in ITS Administrative Controls Section.
3.9.6/4.10.5	COMPLIANCE WITH 40 CFR PART 190 - RADIOACTIVE EFFLUENTS FROM URANIUM FUEL CYCLE SOURCES	None	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-32. In accordance with Generic Letter 89-01, Programmatic Controls are included in ITS Administrative Controls Section.
	REQUIRED SHUTDOWN MARGINS, CONTROL RODS, AND POWER DISTRIBUTION LIMITS				
3.10.1	FULL LENGTH CONTROL ROD INSERTION LIMITS	3.1.3.5 3.1.3.6	3.1.5 3.1.6 3.1.8	Yes-2	The shutdown bank and control bank rod insertion limits functions to ensure the availability of sufficient negative reactivity to shutdown the reactor and maintain the required shutdown margin.
3.10.2	POWER DISTRIBUTION LIMITS	3.2.2 3.2.3 3.2.1	3.2.1 3.2.2 3.2.3	Yes-2	The core power distribution limits function to preclude core power distributions that violate fuel design criteria.
3.10.3	QUADRANT POWER TILT LIMITS	3.2.4	3.2.4	Yes-2	The core power distribution limits function to preclude core power distributions that could result in violation of fuel design criteria.
3.10.4	ROD DROP TIME	3.1.3.4	3.1.4	Yes-2	The Rod Drop Times function to ensure the reactor can be rapidly shutdown consistent with the safety analysis following design basis events.

CURRENT TS NUMBER	DESCRIPTION	NUREG-0452 Rev. 4	ITS NUMBER	RETAINED-CRITERIA FOR INCLUSION	NOTES
3.10.5	REACTOR TRIP BREAKERS	3.3.1	3.3.1	Yes-3	The Reactor Trip Breakers function to interrupt power to permit the control rods to fall into the reactor core.
3.10.6	INOPERABLE CONTROL RODS	3.1.3	3.1.4	Yes-2	Limitations on Control Rod inoperability function to ensure that fuel design limits and RCS integrity are not jeopardized and the core remains subcritical after design basis events.
3.10.7	POWER RAMP RATE LIMITS	None	Relocated	No	See Appendix A page A-34.
3.10.8	REQUIRED SHUTDOWN MARGINS	3.1.1.1 3.1.1.2 3.9.1	3.1.1 3.1.1 3.9.1	Yes-2	The SDM requirements for MODES 1 - 5 function to ensure acceptable fuel design limits are not exceeded for design basis events with the assumption of the highest worth rod stuck out of the core. The SDM requirement for refueling operations function to ensure the reactivity condition of the core is consistent with the applicable safety analysis and is conservative for MODE 6.
3.11	MOVABLE IN-CORE INSTRUMENTATION	3.3.3.2	Relocated	No	See Appendix A page A-36.
3.12	SEISMIC SHUTDOWN	None	Deleted	No	Deleted, see Seismic Shutdown technical change discussion in the Discussion of Changes for CTS 3.12.
3.13	SHOCK SUPPRESSORS (SNUBBERS)	3.7.9	Deleted	No	Deleted, see Snubbers technical change discussion in the Discussion of Changes for CTS 3.13.
3.14	DELETED				
3.15	CONTROL ROOM AIR CONDITIONING	3.7.7	3.7.9 3.7.10	Yes-3	The Control Room Filter System functions to provide a protected environment from which operators can control the unit following an uncontrolled release of radioactive material and to maintain the control room temperature for occupancy following isolation of the control room.
	RADIOACTIVE WASTE SYSTEMS				
3.16.1/4.20.1	LIQUID RADWASTE TREATMENT SYSTEM	None	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-38. In accordance with Generic Letter 89-01, Programmatic Controls are included in ITS Administrative Controls Section.
3.16.2/4.20.2	LIQUID HOLDUP TANKS	None	5.5.12	No	Although this specification does not meet any Technical Specification selection criteria it is retained in accordance with the NRC letter from W. T. Russell to the industry ITS chairperson, dated October 25, 1993.
3.16.3/4.20.3	GASEOUS RADWASTE AND VENTILATION EXHAUST TREATMENT SYSTEMS	None	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-40. In accordance with Generic Letter 89-01, Programmatic Controls are included in ITS Administrative Controls Section.

CURRENT TS NUMBER	DESCRIPTION	NUREG-0452 Rev. 4	ITS NUMBER	RETAINED-CRITERIA FOR INCLUSION	NOTES
3.16.4/4.20.4	WASTE GAS DECAY TANKS (HYDROGEN AND OXYGEN)	None	5.5.12	No	Although this specification does not meet any Technical Specification selection criteria it is retained in accordance with the NRC letter from W. T. Russell to the industry ITS chairperson, dated October 25, 1993.
3.16.5/4.20.5	WASTE GAS DECAY TANKS (RADIOACTIVE MATERIALS)	None	5.5.12	No	Although this specification does not meet any Technical Specification selection criteria it is retained in accordance with the NRC letter from W. T. Russell to the industry ITS chairperson, dated October 25, 1993.
3.16.6/4.20.6	SOLIDIFICATION OF WET RADIOACTIVE WASTE	None	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-42.
	RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM				
3.17.1/4.21.1	MONITORING PROGRAM	None	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-44. In accordance with Generic Letter 89-01, Programmatic Controls are included in ITS Administrative Controls Section.
3.17.2/4.21.2	LAND USE CENSUS	None	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-46. In accordance with Generic Letter 89-01, Programmatic Controls are included in ITS Administrative Controls Section.
3.17.3/4.21.3	INTERLABORATORY COMPARISON PROGRAM	None	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A page A-48. In accordance with Generic Letter 89-01, Programmatic Controls are included in ITS Administrative Controls Section.
4.0	SURVEILLANCE	N/A	N/A	N/A	Application of TS Selection Criteria is not appropriate. However Surveillance requirements are included in ITS required by 10 CFR 50.36. CTS section 4.0 review identified an LCO like requirement. See 4.6.2.
4.6.2	DIESEL FUEL TANKS	3.8.1.1.b	3.8.3	Yes-3	This specification is an LCO located in section 4 of the CTS. The Diesel Fuel subsystem supports the operation of the diesel generators.
4.16	RADIOACTIVE SOURCE LEAKAGE TESTING	3.7.10	Relocated	No	See Appendix A Page A-50
5.0	DESIGN FEATURES	4.0	4.0	Yes	Application of TS Selection Criteria is not appropriate. However Design Features are included in ITS as required by 10 CFR 50.36. CTS Section 5.0 review identified an LCO like requirement. See 5.4.2 and 5.4.3.

CURRENT TS NUMBER	DESCRIPTION	NUREG-0452 Rev. 4	ITS NUMBER	RETAINED-CRITERIA FOR INCLUSION	NOTES
5.4.2.1		None	3.7.14 and 4.3.1	Yes-2	Restrictions on higher enrichment fuel assemblies ensure Keff is maintained <0.95.
5.4.2.2		None	3.7.14	Yes-2	Restrictions on higher enrichment fuel assemblies ensure Keff is maintained <0.95.
5.4.3		None	3.7.13	Yes-2	Restrictions on higher enrichment fuel assemblies ensure Keff is maintained <0.95.
6.0	ADMINISTRATIVE CONTROLS	6.0	5.0	Yes	Application of TS Selection Criteria is not appropriate. However Administrative Controls are included in ITS as required by 10 CFR 50.36
N/A	APPENDIX B	N/A	Relocated	No	This specification does not satisfy selection criteria and is relocated to a licensee controlled document. See Appendix A.page A-52.

APPENDIX A

JUSTIFICATION FOR

SPECIFICATION RELOCATION

3.1.1.4 REACTOR COOLANT SYSTEM (RCS) VENT PATH

LCO Statement:

A. When the RCS temperature is greater that 200°F, the RCS vent paths consisting of at least two valves in series powered from emergency buses, shall be operable (except that valves RC-567, 568, 569 and 570 shall be closed with power removed from the valve actuators) from each of the following locations:

1. Reactor Vessel Head
2. Pressurizer Steam Space

Discussion:

The RCS vent paths are provided to exhaust non-condensible gases and/or steam from the RCS which could inhibit natural circulation core cooling following any event involving a loss of offsite power and requiring long term cooling, such as a Loss-of-Coolant Accident (LOCA). Their function, capabilities, and testing requirements are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," however, the operation of reactor vessel head vents is not assumed in the safety analysis. This is because the operation of the vents is not part of the primary success path in the Updated Final Safety Analysis Report (UFSAR). The operation of these vents is an operator action after the event has occurred, and is only required when there is indication that natural circulation is not occurring.

Comparison to Selection Criteria:

1. RCS vent paths are not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. RCS vent paths are not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. RCS vent paths are not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-44) and summarized in Table 1 of WCAP-11618, the RCS vent paths were found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. RCS vent paths are not important for any scenarios modeled in the HBRSEP, Unit No. 2 Probabilistic Safety Assessment (PSA).

Conclusion:

Since the selection criteria have not been satisfied, the RCS Vent Paths LCO and Surveillances may be relocated to other plant controlled documents outside the TS.

3.1.2.2 STEAM GENERATOR PRESSURE/TEMPERATURE (P/T) LIMITS

LCO Statement:

The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the primary and vessel is below 70°F.

Discussion:

The limitation on steam generator pressures and temperature (i.e., P/T) ensures that pressure-induced stresses on the steam generators do not exceed the maximum allowable fracture toughness limits. These pressure and temperature limits are based on maintaining steam generator RT_{NDT} sufficient to prevent brittle fracture. As such, the TS places limits on variables consistent with structural analysis results. However, these limits are not initial condition assumptions of a UFSAR accident analysis. These limits represent operating restrictions and Criterion 2 includes operating restrictions. However, the Final Policy Statement Criterion 2 discussion specified only those operating restrictions required to preclude unanalyzed accidents and transients be included in TS.

Comparison to Selection Criteria:

1. The steam generator P/T limits are not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Steam generator P/T limits are not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Steam generator P/T limits are not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-55) and summarized in Table 1 of WCAP-11618, the steam generator P/T limits were found to be non-significant risk contributors to core damage frequency and offsite releases. This is, in large part, due to Steam Generator Tube Rupture (SGTR) events being negligible contributors in past PWR PRAs. For HBRSEP, SGTR sequences are important in the HBRSEP PSA. However, this plant-specific PSA does not evaluate conditions below 70°F. In addition, it is also recognized that the likelihood of pressurizing the SG secondary side when RCS temperature is below 70°F is small.

Conclusion:

[illegible]e the selection criteria have not been satisfied, the steam generator P/T limits LCO and Surveillances may be relocated to other plant controlled documents outside the TS.

3.1.2.3 PRESSURIZER TEMPERATURE LIMITS

LCO Statement:

The pressurizer shall neither exceed a maximum heatup rate of 100°F/hr nor a cooldown rate of 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

Discussion:

Limits are placed on pressurizer operation to prevent a non-ductile failure. These limitations are consistent with structural analysis results. However, these limits are not an initial condition assumption of a DBA or transient. These limits represent operating restrictions and Criterion 2 includes operating restrictions. The Final Policy Statement discussion for Criterion 2 specified only those operating restrictions required to preclude unanalyzed accidents and transients be included in Technical Specifications.

Comparison to Selection Criteria:

1. The pressurizer temperature limits are not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. The pressurizer temperature limits are not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The pressurizer temperature limits are not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-41) and summarized in Table 1 of WCAP-11618, the pressurizer temperature limits were found to be non-significant risk contributors to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. The pressurizer temperature limits are outside the scope of the HBRSEP, Unit No.2 PSA, and therefore, the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the pressurizer temperature limits LCO and Surveillances may be relocated to other plant controlled documents outside the TS.

3.1.6, (TABLE 4.1.2, Item 1) MAXIMUM REACTOR COOLANT OXYGEN AND CHLORIDE CONCENTRATION

LCO Statement:

3.1.6.1 The concentration of oxygen in the reactor coolant shall not exceed 0.1 ppm, when the reactor coolant temperature exceeds 250°F.

3.1.6.2 The concentration of chloride in the reactor coolant shall not exceed 0.15 ppm, when the reactor coolant temperature exceeds 250°F.

3.1.6.3 If the oxygen concentration or the chloride concentration of the reactor coolant exceed the limits given in 3.1.6.1 or 3.1.6.2 respectively, corrective action is to be taken immediately to return the system to within normal operation specifications. If the normal operational limits are not achieved within 24 hours, the reactor is to be placed in the cold shutdown condition utilizing normal operating procedures.

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 RCS pressure boundary caused by corrosion. However, the chemistry monitoring activity is of a long term preventative purpose rather than mitigative.

Comparison to Selection Criteria:

1. Reactor coolant water chemistry is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Reactor coolant water chemistry is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Reactor coolant water chemistry is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-40) and summarized in Table 1 of WCAP-11618, the reactor coolant water chemistry was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. Effects of RCS chemistry are outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore, the

plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the RCS Chemistry LCOs and Surveillances may be relocated to other plant controlled documents outside the TS.

3.3.5 POST ACCIDENT CONTAINMENT VENTING SYSTEM

LCO Statement:

3.3.5 The reactor shall not be made critical unless the valves of the post accident containment venting system are operable.

Discussion:

The containment venting system ensures that hydrogen concentration within containment will be maintained below its flammability limit during post LOCA conditions. The containment venting system is capable of controlling expected hydrogen generation associated with: (1) zirconium-water reaction, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. The buildup of hydrogen is expected to be quite small initially, such that the use of the Containment Venting System is not anticipated before 24 hours after the initiation of an accident.

Comparison to Selection Criteria:

1. The containment venting system is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. The containment venting system is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The containment venting system is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-52) and summarized in Table 1 of WCAP-11618, the containment venting system was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. Effects of the containment venting system are outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore, the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the containment venting system LCOs and Surveillances may be relocated to other plant controlled documents outside the TS.

3.5.1 OPERATIONAL SAFETY INSTRUMENTATION

Table 3.5-5

LCO Statement:

3.5.1.2 For on-line testing or in the event of a subsystem instrumentation channel failure, plant operation at rated power shall be permitted to continue in accordance with Tables 3.5-2 through 3.5-5.

3.5.1.3 In the event the number of channels in a particular subsystem in service falls below the limits given in the column entitled Minimum Operable Channels, or Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirements shown in Column 3 of tables 3.5-2 through 3.5-4 and column 2 of Table 3.5-5

Note: *The current TS do not include a unique LCO which requires the operability of the instrumentation identified on Table 3.5-5.*

Discussion:

Each individual accident monitoring parameter has a specific purpose, however, the general purpose for accident monitoring instrumentation is to provide sufficient information to confirm an accident is proceeding as predicted (i.e., automatic safety systems are performing properly, and deviations from expected accident course are minimal).

Comparison to Deterministic Selection Criteria:

The NRC position on application of the deterministic selection criteria to post-accident monitoring instrumentation is documented in NRC letter dated May 9, 1988 from T. E. Murley (NRC) to R. A. Newton (Westinghouse Owners Group). The position was that the post-accident monitoring instrumentation table list should contain, on a plant specific basis, Regulatory Guide 1.97 Type A instruments specified in the plant's Safety Evaluation Report (SER) on Regulatory Guide 1.97, and Regulatory Guide 1.97 Category 1 instruments. Accordingly, this position has been applied to the HBRSEP Regulatory Guide 1.97 instruments. Those instruments meeting this criteria have remained in TS. The instruments not meeting this criteria may be relocated from the TS to plant controlled documents.

The following summarizes the HBRSEP, Unit No. 2 position for those instruments currently in TS:

From NRC SER dated March 5, 1987
Subject: Regulatory Guide 1.97, Emergency Response Capability.

Type A Variables

1. Pressurizer Level
2. Containment Vessel Level (Wide Range)

3. Containment Vessel Pressure (Wide range)
4. Containment Vessel Hydrogen Concentration
5. Incore Thermocouple

Other Type, Category 1 Variables

1. Containment Area Radiation (High Range)

Additional Instrumentation, Associated With Risk Significant Scenarios or Mitigation Systems

(These indications are not specifically modeled in the HBRSEP PSA; however they provide information to the operators regarding risk significant systems modeled in the PSA.)

1. Auxiliary Feedwater Flow (SD AFW Pump)
2. Auxiliary Feedwater Flow (MD AFW Pump)
3. PORV Position Indicator (Primary)
4. PORV Blocking Valve Position Indicator (Primary)
5. Safety Valve Position Indicator

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

Conclusion:

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

1. Reactor Coolant System Subcooling Monitor
2. Noble Gas Effluent Monitor - Main Steam Lines
3. Noble Gas Effluent Monitor - Main Vent Stack High Range
4. Noble Gas Effluent Monitor - Main Vent Stack Mid Range
5. Noble Gas Effluent Monitor - Spent Fuel Pit-Lower Level High Range
6. Reactor Vessel Level Instrumentation System (RVLIS)

3.5.2, (Table 3.5.6)/ 4.19.1 — RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION SYSTEM

LCO Statement:

3.5.2.1 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.5-6 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.9.1.1 are not exceeded. The alarm/trip setpoints shall be determine in accordance with the ODCM.

3.5.2.2 With a radioactive liquid monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, without delay suspend the release of radioactive liquid effluent monitored by the affected channel, change the setpoint so it is acceptably conservative, or declare the channel not operable.

3.5.2.3 With less than the minimum number of radioactive liquid effluent monitoring instrumentation operable, take the action shown in Table 3.5-6.

3.5.2.4 The provisions of Specification 3.0 are not applicable.

Discussion:

The purpose of the Radioactive Liquid Effluent Instrumentation is to monitor routine radioactive releases. This instrumentation provides a surveillance of release points and initiates automatic alarm and trip functions to terminate the release prior to exceeding the limits of 10 CFR 20. The alarm and trip functions are set in accordance with the Offsite Dose Control Manual (ODCM). Radioactive liquid effluent instrumentation and associated requirements for effluent releases are used to assure conformance to the discharge limits of 10 CFR Part 20. The radioactive liquid effluent monitors are used routinely to provide a continuous check on the release of radioactive liquid effluent from the normal plant effluent flow paths. These requirements ensure the various liquid effluent monitors are maintained operable with setpoints established in accordance with the Offsite Dose Calculation Manual (ODCM). Plant DBA and transient analyses do not assume any action, either automatic or manual, resulting from radioactive liquid effluent monitors.

Comparison to Selection Criteria:

1. Radioactive liquid effluent instrumentation is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Radioactive liquid effluent instrumentation is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Radioactive liquid effluent instrumentation is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a

UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, radioactive liquid effluent instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. Effects of radioactive liquid effluent instrumentation are outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore, the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, radioactive liquid effluent instrumentation LCOs may be relocated to other plant controlled documents outside the TS. Consistent with agreements reached with the Industry and NRC, programmatic aspects of this specification are retained as part of a program in ITS Administrative Controls.

3.5.3, (Table 3.5.7)/ 4.19.2 RADIOACTIVE GASEOUS EFFLUENT INSTRUMENTATION SYSTEM

LCO Statement:

3.5.3.1 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.5-7 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.9.3.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.

3.5.3.2 With a radioactive effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, without delay suspend the release of radioactive gaseous effluents, change the setpoint so it is acceptably conservative, or declare the channel not operable.

3.5.3.3 With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels operable take the action shown in Table 3.5-7.

3.5.3.4 The provisions of Specification 3.0 are not applicable.

Discussion:

The purpose of the Radioactive Gaseous Effluent Instrumentation is to monitor routine and control, as applicable, radioactive releases. This instrumentation provides a surveillance of release points and initiates automatic alarm/trip functions to terminate the release prior to exceeding the limits of 10 CFR 20. The alarm/trip functions are set in accordance with the ODCM. Radioactive gases effluent monitoring instrumentation and associated requirements for gaseous effluent releases are used to assure conformance to the discharge limits of 10 CFR Part 20. The radioactive gaseous effluent monitors are used routinely to provide a continuous check on the release of radioactive gaseous effluents from the normal plant gaseous effluent flow paths. These requirements ensure the various effluent monitors are maintained operable with setpoints established in accordance with the Offsite Dose Calculation Manual (ODCM). Plant DBA and transient analyses do not assume any action, either automatic or manual, resulting from radioactive gaseous effluent monitors.

Comparison to Selection Criteria:

1. Radioactive gaseous effluent instrumentation is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Radioactive gaseous effluent instrumentation is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Radioactive gaseous effluent instrumentation is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a

UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, radioactive gaseous effluent instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. Effects of radioactive gaseous effluent instrumentation are outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore, the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, radioactive gaseous liquid effluent instrumentation LCOs may be relocated to other plant controlled documents outside the TS. Consistent with agreements reached with the Industry and NRC, programmatic aspects of this specification are retained as part of a program in ITS Administrative Controls.

3.8.1.c SPENT FUEL STORAGE AREA RADIATION MONITORING DURING REFUELING

LCO Statement:

Radiation levels in the containment and spent fuel storage areas shall be monitored continuously.

Note: This evaluation addresses only the portion of the specification associated with *spent fuel storage area radiation monitoring* being relocated to licensee controlled documents. The portion associated with *containment radiation monitoring* is retained in the HBRSEP, Unit No. 2 ITS.

Discussion:

These radiation monitors do not automatically initiate the Spent Fuel Building Ventilation System. The Spent Fuel Building Ventilation System is required to be in operation with the exhaust discharging through HEPA and impregnated charcoal filters during movement of irradiated fuel in the Spent Fuel Building. Radiation monitoring in the spent fuel storage area during refueling operations provides warning of abnormal or unusually high radiation levels in the spent fuel storage area.

Comparison to Selection Criteria:

1. Radiation monitoring during refueling is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Radiation monitoring during refueling is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Radiation monitoring during refueling is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. Radiation monitoring during refueling is not addressed in WCAP-11618. No HBRSEP PSA risk measure or insight indicates the spent fuel building radiation monitoring LCO is significant to public health or safety. The HBRSEP PSA addresses core damage and radioactive release risk from internal events that are postulated to occur at full power operation. The fire PRA performed for the HBRSEP IPEEE addresses core damage risk from fires at full power operation. Specification 3.8.1c involves a system parameter (spent fuel pool area radiation) and plant mode (refueling) that are not modeled in the Robinson PSA. WCAP-11618, Table 3 identifies Dominant Accident Sequences for plants with a large dry containment. This specification is not directly associated with any of these sequences. WCAP-11618, Tables 3A and 3B identify systems for plants with a large dry

containment where the system's failure contributes to a coremelt frequency > 10^{-6} per year or plant risk > 10^{-7} per year. This specification is not directly associated with plant systems on either table. Consequently, this specification is not considered to be risk significant.

Conclusion:

Since the selection criteria have not been satisfied, the radiation monitoring during refueling LCO for the spent fuel storage area may be relocated to other plant controlled documents outside the TS.

3.8.1.g COMMUNICATIONS DURING REFUELING

LCO Statement:

Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

Discussion:

Communication between the control room personnel and personnel performing core alterations is maintained to ensure that personnel can be promptly informed of significant changes in the plant status or core reactivity condition during refueling. The communications allow for coordination of activities that require interaction between the control room and containment personnel. However, the refueling system design accident or transient response does not take credit for communications in analyzing accident consequences.

Comparison to Selection Criteria:

1. Communications during refueling operations is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Communications during refueling operations is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Communication during refueling operations is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, Page A-67) and summarized in Table 1 of WCAP-11618, the loss of communications was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. The HBRSEP does not cover shutdown conditions and therefore, provides no information to supplement the conclusions of the generic analysis.

Conclusion:

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

3.8.3 SPENT FUEL POOL WATER TEMPERATURE

LCO Statement:

During the discharge of a full core into the spent fuel pit, the temperature of the spent fuel pool water shall be maintained at or below 150°F. The spent fuel pool water temperature shall be monitored once each shift when the temperature is at of below 125°F. If the temperature exceeds 125°F, it shall be monitored hourly. If the pool temperature reaches 150°F, fuel assemblies will be transferred back to the containment to reduce the pool temperature below 150°F.

Discussion:

The spent fuel cooling system is designed to maintain the pool temperature less than or equal to 166°F. The restriction of 150°F provides a margin to prevent the fuel pool temperature from reaching the design value. Plant operating procedures provide adequate controls for this plant parameter. These limits are not related to protection of the public from the consequences of any DBA or transient.

Comparison to Selection Criteria:

1. Spent fuel pool water temperature is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Spent fuel pool water temperature is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Spent fuel pool water temperature is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. Spent fuel pool water temperature is not addressed in WCAP-11618. No HBRSEP PSA risk measure or insight indicates the spent fuel water temperature LCO is significant to public health or safety. The HBRSEP PSA addresses core damage and radioactive release risk from internal events that are postulated to occur at full power operation. The fire PRA performed for the HBRSEP IPEEE addresses core damage risk from fires at full power operation. Specification 3.8.3 involves a system parameter (spent fuel pool temperature) and plant mode (full core discharge) that are not modeled in the Robinson PSA. WCAP-11618, Table 3 identifies Dominant Accident Sequences for plants with a large dry containment. This specification is not directly associated with any of these sequences. WCAP-11618, Tables 3A and 3B identify systems for plants with a large dry containment where the system's failure contributes to a coremelt frequency $> 10^{-6}$ per year or plant risk $> 10^{-7}$ per year. This specification is not directly associated with plant systems on

either table. Consequently, this specification is not considered to be risk significant.

Conclusion:

Since the selection criteria have not been satisfied, the spent fuel pool water temperature LCO and Surveillances may be relocated to other plant controlled documents outside the TS.

3.8.4 SPENT FUEL CASK HANDLING CRANE

LCO Statement:

3.8.4 The following restrictions and requirements shall be applied to the Spent Fuel Cask Handling Crane:

a. Use of the Spent Fuel Cask Handling Crane for lifting operations shall be permitted only when the ambient outside air temperature is greater than 33°F. If the temperature falls below this limit, lifting operations shall be suspended, with the load placed in a safe configuration, until the temperature increases above the limit.

b. Limit switches provided to limit travel of the bridge, trolley, and hoist shall be tested every six months when the crane is not in service, and shall be tested prior to each period of service and on a monthly basis while the crane is in service.

c. Crane ropes shall be inspected in accordance with ANSI B30.2.0 - 1967 every six months when the crane is not inservice, and shall be inspected prior to each period of service and on a monthly basis while the crane is in service. A crane rope shall be replaced if any of the replacement criteria given in ANSI B30.2.0-1967 are met.

Discussion:

The requirement of Specification 3.8.4 are based on limiting the potential for a cask drop accident. Provisions have been made to reduce the potential for a spent fuel cask drop as a credible accident. Redundancy has been incorporated in the design of the spent fuel cask lifting yoke and the 125-ton spent fuel cask handling crane to reduce the risk to public health and safety. A discussion of the safety features of the cask and handling components is contained in Letter NG-74-1246, dated October 17, 1974, CP&L to U.S. Atomic Energy Commission, Spent Fuel Cask Handling. Other actions implemented associated with the control of heavy loads include training of personnel; use of appropriate load handling procedures; identification and utilization of safe load paths; inspection, testing and maintenance of cranes; appropriate design of crane and lifting devices, etc. These actions and the redundancy associated with the design of the spent fuel cask lifting yoke and crane provide reasonable assurance regarding the limited potential for a cask drop accident.

Comparison to Selection Criteria:

1. Spent Fuel Cask Handling Crane restrictions are not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Spent Fuel Cask Handling Crane restrictions are not process variables, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Spent Fuel Cask Handling Crane restrictions are not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. Spent Fuel Cask Handling Crane restrictions are not addressed in WCAP-11618. No HBRSEP PSA risk measure or insight indicates the Spent Fuel Cask Handling Crane restrictions are significant to public health or safety. The HBRSEP PSA addresses core damage and radioactive release risk from internal events that are postulated to occur at full power operation. The fire PRA performed for the HBRSEP IPEEE addresses core damage risk from fires at full power operation. Specification 3.8.4 involves a component (spent fuel handling crane) and plant mode (fuel handling) that are not modeled in the Robinson PSA. WCAP-11618, Table 3 identifies Dominant Accident Sequences for plants with a large dry containment. This specification is not directly associated with any of these sequences. WCAP-11618, Tables 3A and 3B identify systems for plants with a large dry containment where the system's failure contributes to a coremelt frequency $> 10^{-6}$ per year or plant risk $> 10^{-7}$ per year. This specification is not directly associated with plant systems on either table. Consequently, this specification is not considered to be risk significant.

Conclusion:

Since the selection criteria have not been satisfied, the Spent Fuel Cask Handling Crane LCO restrictions may be relocated to other plant controlled documents outside the TS.

3.9.1/4.10.1 COMPLIANCE WITH 10 CFR 20 - RADIOACTIVE MATERIALS IN LIQUID EFFLUENTS

LCO Statement:

3.9.1.1 The concentration of radioactive material in liquid effluents released at any time from the site to unrestricted areas (see Figure 1.1-1) shall be limited to the concentrations specified in 10CFR20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2 x 10-4 μCi/ml total activity.

3.9.1.2 With the concentration of radioactive material in liquid effluents released from the site to unrestricted areas exceeding the above limits, without delay restore the concentration to within the above limits. In addition, notification must be made to the Commission in accordance with Specification 6.6.

3.9.1.3 In the event that the immediate action required by 3.9.1.2 above cannot be satisfied, the facility shall be placed in hot shutdown within 12 hours and in cold shutdown within the next 30 hours, and entry into the power operating condition shall not be made unless Specification 3.9.1.1 is met.

3.9.1.4 The provisions of Specification 3.0 are not applicable.

Discussion:

The Liquid Effluent Concentration Limit ensures that the concentration of radioactive materials released in liquid waste effluent to unrestricted areas will be less than the concentration levels specified in 10 CFR 20, Appendix B. 10 CFR Part 20, BII(2) refers to liquid release to an unrestricted area of radioactive material in concentrations that exceed the specified limits. No screening criteria apply because the process variable of the LCO (concentration of radioactive material in liquid effluents) is not an initial condition of a design basis accident (DBA) or transient analysis. Effluent control is for protection against radiation hazards from licensed activities, not accidents.

Comparison to Selection Criteria:

1. Compliance with 10 CFR 20 - Radioactive Materials in Liquid Effluents is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Compliance with 10 CFR 20 - Radioactive Materials in Liquid Effluents is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Compliance with 10 CFR 20 - Radioactive Materials in Liquid Effluents is not a structure, system, or component that is part of the primary success path and functions

or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, Compliance with 10 CFR 20 - Compliance with 10 CFR 20 - Radioactive Materials in Liquid Effluents was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. Effects of Compliance with 10 CFR 20 - Radioactive Materials in Liquid Effluents are outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore, the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the 10 CFR 20 - Radioactive Materials in Liquid Effluents LCO and Surveillances may be relocated to other plant controlled documents outside the TS. Consistent with agreements reached with the Industry and NRC, programmatic aspects of this specification are retained as part of a program in ITS Administrative Controls.

3.9.2 COMPLIANCE WITH 10 CFR 50 RADIOACTIVE MATERIALS IN LIQUID EFFLUENTS

LCO Statement:

3.9.2.1 The dose commitment at all times to a member of the public from radioactive materials in liquid effluents released to unrestricted areas (See Figure 1.1-1) shall be limited:

a. During any calendar quarter to ≤ 1.5 mrem to the total body and to ≤ 5 mrem to any organ, and

b. During any calendar year to ≤ 3 mrem to the total body and to ≤ 10 mrem to any organ.

3.9.2.2 With the calculated dose commitment from the release of radioactive materials in liquid effluents exceeding any of the limits prescribed by Specification 3.9.2.1 above, prepare and submit a report to the Commission in accordance with Specification 6.9.3.2.

Discussion:

This specification is provided to implement the requirements of Sections II.A, and III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The action statement provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I of 10 CFR Part 50 to assure that the release of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." Limitation of the quarterly and annual projected doses to MEMBERS OF THE PUBLIC which result from cumulative liquid effluent discharge during normal operation over extended periods is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I. These limits are not related to protection of the public from the consequences of any DBA or transient.

Comparison to Selection Criteria:

1. Compliance with 10 CFR 50 - Radioactive Materials in Liquid Effluents is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Compliance with 10 CFR 50 - Radioactive Materials in Liquid Effluents is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Compliance with 10 CFR 50 - Radioactive Materials in Liquid Effluents is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, Compliance with 10 CFR 50 - Radioactive Materials in Liquid Effluents was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. Effects of Compliance with 10 CFR 50 - Radioactive Materials in Liquid Effluents are outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore, the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the Compliance with 10 CFR 50 - Radioactive Materials in Liquid Effluents LCO and Surveillances may be relocated to other plant controlled documents outside the TS. Consistent with agreements reached with the Industry and NRC, programmatic aspects of this specification are retained as part of a program in ITS Administrative Controls.

3.9.3/4.10.2 COMPLIANCE WITH 10 CFR 20 - RADIOACTIVE MATERIAL IN GASEOUS EFFLUENTS

LCO Statement:

3.9.3.1 The dose rate due to radioactive materials in gaseous effluents released from the site boundary (see Figure 1.1-1) shall be limited to the following:

a. For radionoble gases: ≤ 500 mrem/yr to the total body, ≤ 3000 mrem/yr to the skin, and

b. For I-131, I-133, and tritium, and for all radioactive materials in particulate form, inhalation pathway only, with half lives greater than 8 days: ≤ 1500 mrem/yr to any organ.

3.9.3.2 With the dose rate(s) exceeding the above limits, without delay decrease the release rate to within the above limits. In addition, a notification must be made to the Commission in accordance with Specification 6.6.

3.9.3.3 In the event that the immediate action required by 3.9.3.2 above cannot be satisfied, the facility shall be placed in hot shutdown within 12 hours and in cold shutdown within the next 30 hours, and entry into the power operating condition shall not be made until Specification 3.9.3.1 is met.

Discussion:

This specification is provided to ensure the dose rate at any time at the site boundary from gaseous effluents is within the annual dose limits of 10 CFR 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR 20 Appendix B, Table II, Column 1. These are limits which apply to normal operation of the plant. They are not assumed as an initial condition of any design basis accident (DBA) or transient analysis and are not relied upon to limit the consequences of such events.

Comparison to Selection Criteria:

1. Compliance with 10 CFR 20 - Radioactive Material in Gaseous Effluents is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Compliance with 10 CFR 20 - Radioactive Material in Gaseous Effluents is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Compliance with 10 CFR 20 - Radioactive Material in Gaseous Effluents is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, Compliance with Compliance with 10 CFR 20 - Radioactive Material in Gaseous Effluents was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. Effects of Compliance with 10 CFR 20 - Radioactive Material in Gaseous Effluents are outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore, the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the Compliance with Compliance with 10 CFR 20 - Radioactive Material in Gaseous Effluents LCO and Surveillances may be relocated to other plant controlled documents outside the TS. Consistent with agreements reached with the Industry and NRC, programmatic aspects of this specification are retained as part of a program in ITS Administrative Controls.

3.9.4/4.10.3 COMPLIANCE WITH 10 CFR 50 - RADIONOBLE GASES

LCO Statement:

3.9.4.1 The air dose commitment due to radionoble gases released in gaseous effluents to areas at and beyond the site boundary (See Figure 1.1-1) shall be limited, at all times, to the following:

a. During any calendar quarter, to ≤ 5 mrad for gamma radiation and ≤ 10 mrad for beta radiation;

b. During any calendar year, to ≤ 10 mrad for gamma radiation and ≤ 20 mrad for beta radiation.

3.9.4.2 With the calculated air dose commitment from radioactive noble gases in gaseous effluents exceeding any of the limits, prescribed by Specification 3.9.4.1 above, prepare and submit a report to the Commission in accordance with Specification 6.9.3.2.

Discussion:

The specification ensures that the concentration of radioactive materials released in gaseous effluent to unrestricted areas are kept as low as reasonably achievable. This specification is provided to implement the requirements of Section II. B, III.A and IV.A of Appendix I, 10 CFR Part 50. The limiting condition for operation implementing the guides provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." These limits are not related to protection of the public from the consequences of any DBA or transient.

Comparison to Selection Criteria:

1. Compliance with 10 CFR 50 - Radionoble Gases is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Compliance with 10 CFR 50 - Radionoble Gases is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Compliance with 10 CFR 50 - Radionoble Gases is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, Compliance with 10 CFR 50 - Radionoble Gases was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L

has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. Effects of Compliance with 10 CFR 50 - Radionoble Gases are outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore, the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the Compliance with 10 CFR 50 - Radionoble Gases LCO and Surveillances may be relocated to other plant controlled documents outside the TS. Consistent with agreements reached with the Industry and NRC, programmatic aspects of this specification are retained as part of a program in ITS Administrative Controls.

3.9.5/4.10.4 COMPLIANCE WITH 10 CFR 50 - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN RADIONOBLE GASES

LCO Statement:

3.9.5.1 The dose to a member of the public from I-131, I-133, tritium and radioactive materials in particulate form, with half-lives greater than 8 days in gaseous effluents released to unrestricted areas (See Figure 1.1-1), shall be limited, at all times, to the following:

a. During any calendar quarter, ≤ 7.5 mrem to any organ and,

b. During any calendar year, ≤ 15 mrem to any organ.

3.9.5.2 With the calculated dose commitment from the release of I-131, I-133, tritium and radioactive materials in particulate form, with half lives greater than 8 days, in gaseous effluents exceeding any of the limits prescribed by Specification 3.9.5.1 above, prepare and submit a report to the Commission in accordance with Specification 6.9.3.2.

Discussion:

This specification is provided to implement the requirements of Section II. C, III.A, and IV.A of Appendix I, 10 CFR Part 50. The limiting condition for operation implements the guides set forth in Section II.C of Appendix I. The action statement provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials as gaseous effluents will be kept "as low as reasonably achievable." These limits are not related to protection of the public from the consequences of any DBA or transient.

Comparison to Selection Criteria:

1. Compliance with 10 CFR 50 - Compliance with 10 CFR 50 - Radioiodines, Radioactive Materials in Particulate Form and Radionuclides other than Radionoble Gases is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Compliance with 10 CFR 50 - Radioiodines, Radioactive Materials in Particulate Form and Radionuclides other than Radionoble Gases is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Compliance with 10 CFR 50 - Radioiodines, Radioactive Materials in Particulate Form and Radionuclides other than Radionoble Gases is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents

a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, Compliance with 10 CFR 50 - Radioiodines, Radioactive Materials in Particulate Form and Radionuclides other than Radionoble Gases was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. Effects of Compliance with 10 CFR 50 - Radioiodines, Radioactive Materials in Particulate Form and Radionuclides other than Radionoble Gases are outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore, the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the Compliance with 10 CFR 50 - Radioiodines, Radioactive Materials in Particulate Form and Radionuclides other than Radionoble Gases LCO and Surveillances may be relocated to other plant controlled documents outside the TS. Consistent with agreements reached with the Industry and NRC, programmatic aspects of this specification are retained as part of a program in ITS Administrative Controls.

3.9.6/4.10.5 COMPLIANCE WITH 10 CFR 190 - RADIOACTIVE EFFLUENT FROM URANIUM FUEL CYCLE SOURCES

LCO Statement:

3.9.6.1 The dose commitment to any member of the public, due to releases of licensed materials and radiation, from uranium fuel cycle sources shall be limited to ≤ 25 mrem to the total body or any organ except the thyroid, which shall be limited to ≤ 75 mrem over 12 consecutive months. This specification is applicable to Robinson Unit 2 only for the area within a five mile radius around the Robinson Plant.

3.9.6.2 With the calculated doses from the release of the radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.9.2.1.a, 3.9.2.1.b, 3.9.4.1.a, 3.9.4.1.b, 3.9.5.1.a, or 3.9.5.1.b, calculations should be made including direct radiation contributions from the reactor unit and from outside storage tanks to determine whether the above limits of Specification 3.9.6.1 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3.2.d, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c shall include an analysis that estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the same request is complete.

3.9.6.3 The provisions of Specification 3.0 are not applicable.

Discussion:

This specification ensures the dose limitations of 10 CFR 40 Part 190 which were incorporated into 10 CFR 20 are not exceeded. This is intended to assure that normal operation of the plant is in compliance with the provisions of 40 CFR Part 190. These limits are not related to protection of the public from any design basis accident (DBA) or transient.

Comparison to Selection Criteria:

1. Compliance with 10 CFR 190 - Radioactive Effluent from Uranium Fuel Cycle Sources is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Compliance with 10 CFR 190 - Radioactive Effluent from Uranium Fuel Cycle Sources is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Compliance with 10 CFR 190 - Radioactive Effluent from Uranium Fuel Cycle Sources is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, Compliance with 10 CFR 190 - Radioactive Effluent from Uranium Fuel Cycle Sources was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. Effects of Compliance with 10 CFR 190 - Radioactive Effluent from Uranium Fuel Cycle Sources are outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore, the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the Compliance with 10 CFR 190 - Radioactive Effluent from Uranium Fuel Cycle Sources LCO and Surveillances may be relocated to other plant controlled documents outside the TS. Consistent with agreements reached with the Industry and NRC, programmatic aspects of this specification are retained as part of a program in ITS Administrative Controls.

3.10.7 POWER RAMP RATE LIMITS

LCO Statement:

3.10.7.1 During the return to power following a shutdown where fuel assemblies have been handled (e.g., refueling, inspection), the rate of reactor power increase shall be limited to 3 percent of rated power in an hour between 20 percent and 100 percent of rated power. This ramp rate requirement applies during the initial startup and may apply during subsequent power increases, depending on the maximum power level achieved and length of operation at that power level. Specifically, this requirement can be moved for reactor power levels below a power level P (20 percent $<$ P $\leq$ 100 percent), provided that the plant has operated at or above power level P for at least 72 cumulative hours out of any seven-day operating period following the shutdown.

The rate of reactor power increases above the highest power level sustained for at least 72 cumulative hours during the preceding 30 cumulative days of reactor power operation shall be limited to 3 percent of rated power in an hour. Alternatively, reactor power increase can be accomplished by a single step increase less than or equal to 10 percent of rated power followed by a maximum ramp rate of 3 percent of rated power in an hour beginning three hours after the step increase.

Discussion:

Calculations show that high cladding stresses can occur if the reactor power increase is rapid after startup from a refueling. The 72 hour period allows for thermal stress relaxation of the clad before the ramp rate requirement is removed, therefore reducing the potential harmful effects of possible pellet or fragment relocation. The 3 percent limit is imposed to minimize the effects of adverse cladding stresses resulting from reduced power operation for extended periods of time. The time period of 30 days is based upon the successful power ramp demonstrations performed on Zircaloy clad in operating reactors, resulting in no cladding failures. The limits associated with this specification are related to minimizing fuel clad damage normal operation. ITS LCO 3.4.16, RCS Specific Activity provides controls to limit allowable radionuclides in the RCS. The limits associated with CTS 3.10.7 are not directly related to a DBA or transient.

Comparison to Selection Criteria:

1. The power ramp rate limits are not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. The power ramp rate limits are not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The power ramp rate limits are not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The power ramp rate limits are not addressed in WCAP-11618. No HBRSEP PSA risk measure or insight indicates the Power Ramp Rate Limit LCO is significant to public health or safety. The HBRSEP PSA addresses core damage and radioactive release risk from internal events that are postulated to occur at full power operation. The fire PRA performed for the HBRSEP IPEEE addresses core damage risk from fires at full power operation. Specification 3.10.7 involves a system parameter (rate of power change) and plant mode (less than full power with power level increasing) that are not modeled in the Robinson PSA. WCAP-11618, Table 3 identifies Dominant Accident Sequences for plants with a large dry containment. This specification is not directly associated with any of these sequences. WCAP-11618, Tables 3A and 3B identify systems for plants with a large dry containment where the system's failure contributes to a coremelt frequency $> 10^{-6}$ per year or plant risk $> 10^{-7}$ per year. This specification is not directly associated with plant systems on either table. Consequently, this specification is not considered to be risk significant.

Conclusion:

Since the selection criteria have not been satisfied, the power ramp rate limits LCO may be relocated to other plant controlled documents outside the TS.

3.11 MOVABLE IN-CORE INSTRUMENTATION

LCO Statement:

3.11.1 A minimum of 15 total accessible thimbles and at least 2 per quadrant sufficient movable in-core detectors shall be operable during recalibration of the excore symmetrical offset detection system.

3.11.2 Power shall be limited to 90% of rated power if recalibration requirements for the excore symmetrical offset detection system defined in Table 4.1-1 are not met.

Discussion:

The movable in-core instrumentation is used to determine the gross power distribution in the core as indicated by the power balance between the top and bottom halves of the core. The full system has more capability than is needed for the calibration of the excore detectors. If the calibration is not performed, the mandated power reduction assures safe operation since it will compensate for an error of 10% in the excore detection system.

Comparison to Selection Criteria:

1. The movable in-core instrumentation is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. The movable in-core instrumentation is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The movable in-core instrumentation is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-12) and summarized in Table 1 of WCAP-11618, the movable in-core instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. Effects of the movable in-core instrumentation are outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the movable in-core instrumentation LCO and Surveillances may be relocated to other plant controlled documents outside the TS.

3.16.1/4.20.1 LIQUID RADWASTE TREATMENT SYSTEM

LCO Statement:

3.16.1.1 The appropriate portions of the Liquid Radwaste Treatment System shall be maintained and used to reduce the concentrations of radioactive materials in liquid wastes prior to their discharge when the projected dose commitments, due to the release of radioactive liquid effluents to unrestricted areas (See Figure 1.1-1) when averaged over a calendar quarter, would exceed 0.2 mrem to the total body or 0.6 mrem to any organ.

3.16.1.2 With radioactive liquid wastes being discharged without treatment while in excess of the limits of Specification 3.16.1.1 above, prepare and submit a report to the Commission in accordance with Specification 6.9.3.2.b.

Discussion:

The Liquid Radwaste Treatment System ensures that effluents will be treated prior to release to the environment. Appropriate portions of the system are required to be operable to maintain doses as low as reasonably achievable. The requirement for a liquid waste treatment system pertains to controlling the release of site liquid effluents during normal operational occurrences. No loss of primary coolant is involved, neither is an accident condition assumed or implied. The limits for release in 10 CFR Part 50, Appendix I for liquids are design objectives for operation. In addition, the liquid radwaste subsystems are not credited in the safety sequence analysis and are not part of the primary coolant pressure boundary.

Comparison to Selection Criteria:

1. The Liquid Radwaste Treatment System is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. The Liquid Radwaste Treatment System is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The Liquid Radwaste Treatment System is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, the Liquid Radwaste Treatment System was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. Effects of the Liquid Radwaste Treatment System are outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore, the plant-specific PSA provides no

information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the Liquid Radwaste Treatment System LCO and Surveillances may be relocated to other plant controlled documents outside the TS. Consistent with agreements reached with the Industry and NRC, programmatic aspects of this specification are retained as part of a program in ITS Administrative Controls.

3.16.3/4.20.3 GASEOUS RADWASTE AND VENTILATION EXHAUST TREATMENT SYSTEMS

LCO Statement:

3.16.3.1 The appropriate portions of the Gaseous Radwaste Treatment System and the Ventilation Exhaust Treatment System shall be maintained and used to reduce the concentrations of radioactive materials in gaseous wastes prior to their discharge when the projected dose commitments due to the release of gaseous effluents to unrestricted areas (See Figure 1.1-1) when averaged over a calendar quarter would exceed:

a. 0.6 mrem for gamma radiation and 1.3 mrem for beta radiation due to radionoble gases or,

b. 1.0 mrem to any organ due to radioiodines, radioactive materials in particulate form, and radionuclides other than radionoble gases.

3.16.3.2 With the Gaseous Radwaste Treatment System and/or the Ventilation Exhaust Treatment System not operable and with radioactive gaseous wastes being discharged without treatment while in excess of the limits of Specification 3.16.3.1 above, prepare and submit a report to the Commission in accordance with Specification 6.9.3.2.b.

Discussion:

The specification ensures that appropriate portions of these systems are maintained and used when specified to ensure that the releases of radioactive material in gaseous effluent is kept as low as reasonably achievable. In addition, the operability of the gaseous radwaste treatment system is not assumed in the analysis or any DBA or transient.

Comparison to Selection Criteria:

1. The Gaseous Radwaste and Ventilation Exhaust Treatment Systems are not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. The Gaseous Radwaste and Ventilation Exhaust Treatment Systems are not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The Gaseous Radwaste and Ventilation Exhaust Treatment Systems are not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, the Gaseous Radwaste and Ventilation Exhaust Treatment Systems

was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. Effects of the Gaseous Radwaste and Ventilation Exhaust Treatment Systems are outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore, the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the Gaseous Radwaste and Ventilation Exhaust Treatment Systems LCO and Surveillances may be relocated to other plant controlled documents outside the TS. Consistent with agreements reached with the Industry and NRC, programmatic aspects of this specification are retained as part of a program in ITS Administrative Controls.

3.16.6/4.20.6 SOLIDIFICATION OF WET RADIOACTIVE WASTE

LCO Statement:

3.16.6.1 The Solid Radwaste System shall be used in accordance with a Process Control Program (PCP) to process wet radioactive waste to meet shipping and burial ground requirements.

3.16.6.2 With the provisions of the PCP not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive waste from the site.

3.16.6.3 If any test specimen, as required by the PCP, fails to verify solidification, the solidification of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative solidification parameters can be determined in accordance with the PCP, and a subsequent test verifies solidification. The PCP shall be modified as required in accordance with Section 6.15, and solidification of the batch may then be resumed using alternative solidification parameters as determined by the PCP.

Discussion:

This specification ensures that the packaging of wet radioactive waste meets the requirements of 10 CFR 20 and 10 CFR 71 prior to their shipment from the site for disposal. The solid radioactive waste system is a logical continuation of the liquid radwaste system. It operates on the same requirement for effluent control, identified as controlling the release and handling of radioactive solid wastes. The system serves to control operational release of solid waste, not accidental release.

Comparison to Selection Criteria:

1. The solidification of wet radioactive waste limits are not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. The solidification of wet radioactive waste limits are not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The solidification of wet radioactive waste limits are not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, the solidification of wet radioactive waste limits were found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. Effects of the solidification of wet radioactive waste limits are outside the scope of the

HBRSEP, Unit No. 2 PSA, and therefore, the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied the solidification of wet radioactive waste limits LCO and Surveillances may be relocated to other plant controlled documents outside the TS.

3.17.1/4.21.1 MONITORING PROGRAM

LCO Statement:

3.17.1.1 The Radiological Environmental Monitoring Program shall be conducted as specified in Table 3.17-1.

3.17.1.2 With the radiological environmental monitoring program not being conducted as specified in Table 3.17-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.e, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

3.17.1.3 With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.17-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a member of the public is less than the calendar year limits of Specifications 3.9.2.1, 3.9.4.1, and 3.9.5.1. When more than one of the radionuclides in Table 3.17-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1\text{-}0$$

When radionuclides other than those in Table 3.17-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose' to a member of the public is equal to or greater than the calendar year limits of Specifications 3.9.2.1, 3.9.4.1, and 3.9.5.1. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

3.17.1.4 With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.17-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.9.1.d, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

3.17.1.5 The provisions of Specification 3.0 are not applicable.

3.17.1.6 Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, or to

malfunction of automatic sampling equipment. If the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.

Discussion:

The Environmental Monitoring Program provides data on radiation levels and radioactive materials in exposure pathways for those radionuclides that lead to the highest potential radiation exposure to members of the public resulting from plant operation. This is accomplished by effluent measurement and modeling the environmental exposure pathways. This program is not related to protection of the public from any design basis accident (DBA) or transient.

Comparison to Selection Criteria:

1. The Environmental Monitoring Program is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. The Environmental Monitoring Program is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The Environmental Monitoring Program is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, The Environmental Monitoring Program was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. The effect of the Environmental Monitoring Program is outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the Environmental Monitoring Program LCO and surveillances may be relocated to other plant controlled documents outside the TS. Consistent with agreements reached with the Industry and NRC, programmatic aspects of this specification are retained as part of a program in ITS Administrative Controls.

3.17.2/4.21.2 LAND USE CENSUS

LCO Statement:

3.17.2.1 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles.

3.17.2.2 With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater that the values currently being calculated in Specification 4.10.4.1, identify the new location(s) in the next Semiannual Radioactive Effluent report, pursuant to Specification 6.9.1.d.

3.17.2.3 With the land use census identifying a location which yields an annual calculated dose or dose commitment of a specific pathway which is 20% greater than that at a current sampling location:

(a) add the new location(s) to the radiological environmental monitoring program within 30 days and,

(b) if desired, delete the sampling location having the lowest calculated dose or dose commitments via the same exposure pathway, excluding the control station location, from the monitoring program after October 31 of the year in which the land use census was conducted, and

(c) identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, Specification 6.9.I.d, including a revised figure(s) and table for the ODCM reflecting the new location(s).

Discussion:

The Land Use Census ensures that changes in the use of land are identified and accounted for in the Radiological Environmental Monitoring Program given in the ODCM. This program is not related to protection of the public from any design basis accident (DBA) or transient.

Comparison to Selection Criteria:

1. The Land Use Census is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. The Land Use Census is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The Land Use Census is not a structure, system, or component that is part of the

primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, The Land Use Census was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. The effect of the Land Use Census is outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the Land Use Census LCO and surveillances may be relocated to other plant controlled documents outside the TS. Consistent with agreements reached with the Industry and NRC, programmatic aspects of this specification are retained as part of a program in ITS Administrative Controls.

3.17.3/4.21.3 INTERLABORATORY COMPARISON PROGRAM

LCO Statement:

3.17.3.1 Analyses shall be performed on radioactive materials supplied by EPA as a part of an Interlaboratory Comparison Program of like media within the environmental program as per Table 3.I7-1.

3.17.3.2 With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.e.

3.17.3.3 The provisions of Specification 3.0 are not applicable.

3.17.3.4 The Interlaboratory comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.e.

Discussion:

The interlaboratory comparison program ensures that independent checks on the precision and accuracy of the measurements of radioactive materials in the environmental samples are performed as part of the quality assurance program for the environmental monitoring program. This program is not related to protection of the public from any design basis accident (DBA) or transient.

Comparison to Selection Criteria:

1. The Interlaboratory Comparison Program is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. The Interlaboratory Comparison Program is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. The Interlaboratory Comparison Program is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, the Interlaboratory Comparison Program was found to be a non-significant risk contributor to core damage frequency and offsite

releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. The effect of the Interlaboratory Comparison Program is outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, the Interlaboratory Comparison Program and surveillances may be relocated to other plant controlled documents outside the TS. Consistent with agreements reached with the Industry and NRC, programmatic aspects of this specification are retained as part of a program in ITS Administrative Controls.

4.16 RADIOACTIVE SOURCE LEAKAGE TESTING

LCO Statement:

The following surveillance requirements imply LCOs exist, however, unique LCOs are not specifically identified in Section 3 of the CTS. These surveillance requirements are comparable to surveillance requirements contained in 3/4.7.10, Sealed Source Contamination of NUREG-0452, Rev. 4, "Standard Technical Specifications For Westinghouse Pressurized Water Reactors."

4.16.1 The leakage test shall be capable of detecting the presence of .005 microcurie of radioactive material on the test sample. If the test reveals the presence of .005 microcurie or more of removable contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or be disposed of in accordance with Commission regulations. Sealed sources are exempt from such leak tests when the source contains 100 microcuries or less of beta and/or gamma emitting material or 10 microcuries or less of alpha emitting material.

4.16.2 Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement State as follows:

A. Each sealed source, except startup sources subject to core flux, containing radioactive material, other than Hydrogen 3, with a half-life greater than thirty days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

B. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer.

In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

C. Startup sources shall be leak tested prior to and following any repair or maintenance and before being subjected to core flux.

Discussion:

This specification ensures that leakage from Byproduct, Source and Special Nuclear Material sources will not exceed allowable intake values. The limitation on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR Part 70.39(a)(3) limits for plutonium. This program is not related to protection of the public from any design basis accident (DBA) or transient.

Comparison to Selection Criteria:

1. Sealed Source Contamination is not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Sealed Source Contamination is not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Sealed Source Contamination is not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, Sealed Source Contamination was found to be a non-significant risk contributor to core damage frequency and offsite releases. CP&L has reviewed this evaluation and considers it applicable to HBRSEP, Unit No. 2. The effect of Sealed Source Contamination is outside the scope of the HBRSEP, Unit No. 2 PSA, and therefore the plant-specific PSA provides no information to supplement the conclusions from the generic analysis.

Conclusion:

Since the selection criteria have not been satisfied, radioactive source leakage testing may be relocated to other plant controlled documents outside the TS.

APPENDIX B (TECHNICAL SPECIFICATION)

LCO Statement:

A. Radioactive Effluent Releases

A statement of the quantities of radioactive effluents released from the plant with data summarized on a monthly basis following the format of USNRC Regulatory Guide 1.21.

1. Gaseous Effluents

(a) Gross Radioactivity Releases

(1) Total gross radioactivity (in curies), primarily noble and activation gases.

(2) Maximum gross radioactivity release rate during any one-hour period.

(3) Total gross radioactivity (in curies) by nuclide released based on representative isotopic analyses performed.

(4) Percent of technical specification limit.

(b) Iodine Releases

(1) Total iodine radioactivity (in curies) by nuclide released based on representative isotopic analyses performed.

(2) Percent of technical specification limit for I-131 released.

(c) Particulate Releases

(1) Total gross radioactivity ($\beta\gamma$) released (in curies) excluding background radioactivity.

(2) Gross alpha radioactivity released (in curies) excluding background radioactivity.

(3) Total gross radioactivity (in curies) of nuclides with half-lives greater than eight days.

(4) Percent of technical specification limit for particulate radioactivity with half-lives greater than eight days.

2. Liquid Effluents

(a) Total gross radioactivity ($\beta\gamma$) released (in curies) excluding tritium and average concentration released to the unrestricted area.

(b) The maximum concentration of gross radioactivity ($\beta\gamma$) released to the unrestricted area (averaged over the period of release).

(c) Total tritium and total alpha radioactivity (in curies) released and average concentration released to the unrestricted area.

(d) Total dissolved gas radioactivity (in curies) and average concentration released to the unrestricted area.

(e) Total volume (in liters) of liquid waste released.

(f) Total volume (in liters) of dilution water used prior to release from the restricted area.

(g) Total gross radioactivity (in curies) by nuclide released based on representative isotopic analyses performed.

(h) Percent of technical specification limit for total radioactivity.

B. Solid Waste

1. The total amount of solid waste shipped (in cubic feet).

2. The total estimated radioactivity (in curies) involved.

3. Disposition including date and destination.

C. Environmental Monitoring

1. For each medium sampled during the reporting period, e.g., air, baybottom, surface water, soil, fish, include:

(a) Number of sampling locations,

(b) Total number of samples,

(c) Number of locations at which levels are found to be significantly above local backgrounds, and

(d) Highest, lowest, and the average concentrations or levels or radiation for the sampling point with the highest average and description of the location of that point with respect to the site.

Discussion:

The Appendix B Technical Specifications contain environmental reporting requirements which were relocated to Appendix B as an interim action in 1976 pending completion of issuance of comprehensive Appendix B Environmental Technical Specifications. These requirements are comparable to portions of other Radiological Environmental Monitoring Technical Specifications which are also being separately relocated.

Comparison to Selection Criteria:

1. Appendix B technical specifications are not installed instrumentation used for, or capable of, detecting and indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

2. Appendix B technical specifications are not a process variable, design feature, or operating restriction that is an initial condition of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. Appendix B technical specifications are not a structure, system, or component that is part of the primary success path and functions or actuates in the mitigation of a UFSAR accident analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. The Appendix B technical specifications are not addressed in WCAP-11618, nor are they modeled in the HBRSEP, Unit No. 2 PSA. WCAP-11618, Table 3 identifies Dominant Accident Sequences for plants with a large dry containment. This specification is not directly associated with any of these sequences. WCAP-11618, Tables 3A and 3B identify systems for plants with a large dry containment where the system's failure contributes to a coremelt frequency $> 10^{-6}$ per year or plant risk $> 10^{-7}$ per year. This specification is not directly associated with plant systems on either table. Consequently, this specification is not considered to be risk significant.

Conclusion:

Since the selection criteria have not been satisfied, the Appendix B technical specifications may be relocated to other plant controlled documents outside the TS. Programmatic aspects of this specification are retained in ITS Administrative Controls.

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MATRIX OF RELOCATED TECHNICAL SPECIFICATIONS (TS) REQUIREMENTS AND DETAILED TS REQUIREMENTS

Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 2.0 LA1	6.7.1.c,d,e Safety limit reporting requirements	Plant Administrative Procedures for Reporting Information to the NRC
ITS 3.0 LA1	4.0 "Prior to returning the system to service, the specified calibration and testing surveillance shall be performed."	Bases of ITS section Surveillance Requirement (SR) 3.0.1.
ITS 3.1 LA1	3.10.8.1 "Shutdown Margin - Hot Shutdown"	Core Operating Limits Report (COLR)
ITS 3.1 LA1	3.10.8.2 "Shutdown Margin - Cold Shutdown"	COLR
ITS 3.1 LA1	3.10.1.4 "At 50% of the cycle as defined by burnup, the limits shall be adjusted to the end-of-core values as specified in the COLR."	COLR
ITS 3.1 LA2	3.10.1.3 "If bank insertion is not restored to the specified limits"..."the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures within six hours."	Reference to "utilizing normal operating procedures" is relocated to plant operating and administrative procedures.
ITS 3.1 R1	3.10.7 Restrictions placed on power ramp rate following a shutdown where core fuel assemblies have been handled	Licensee Controlled Documents (LCDs)

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.2 LA1	3.10.2.1 Algorithms describing the limits of $F_Q(Z)$	COLR
ITS 3.2 LA1	3.10.2.2 Algorithms describing the limits of $F_Q(Z)$ with core penalty factor, V(Z), included	COLR
ITS 3.2 LA2	3.10.2.1 Algorithm describing the limits of $F_{\Delta}H$, uncertainty factor, and power factor Multiplier	COLR
ITS 3.2 LA3	3.10.2.11 Details concerning the redefinition of the axial flux target bands	COLR
ITS 3.2 LA4	3.10.2.2.2 Details concerning the description of determining $F_Q(Z)$ from a power distribution map in terms of measurement and engineering factor uncertainties, and Allowable Power Level	Bases of ITS Section 3.2.1 and 3.2.3
ITS 3.3 LA1	2.3.1.2.d Details concerning how the electronic dynamic compensation and delta flux input to the Overtemperature ΔT Reactor Protection System (RPS) function affects its setpoint	Bases of ITS Section 3.3.1.

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MATRIX OF RELOCATED TECHNICAL SPECIFICATIONS (TS) REQUIREMENTS AND DETAILED TS REQUIREMENTS

Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.3 LA1	2.3.1.2.d Details concerning how the electronic dynamic compensation and delta flux input to the Overpower ΔT RPS function affects its setpoint	Bases of ITS Section 3.3.1.
ITS 3.3 LA2	2.3.3 "The RCS narrow range temperature sensors response time shall be less than or equal to a 4.0 second lag constant."	Bases of ITS section SR 3.3.1.12 with specific time constant specifications relocated to the LCDs.
ITS 3.3 LA3	3.10.5.1.b "The reactor shall not be made critical"..."unless the reactor trip bypass breakers are racked out or removed."	LCDs
ITS 3.3 LA4	Table 4.1-1, Instrument channel Surveillance Requirements.	LCDs
ITS 3.3 LA5	Table 3.5-1, Item 7, Containment monitor, alarm setpoint methodology	OCDM
ITS 3.3 LA6	Table 3.5-5, Note 5, Preplanned alternate method of monitoring be available	LCDs
ITS 3.3 LA7	Table 3.5-2 Item 15A. Control Rod Misalignment Monitor function as provided by the "ERFIS Rod Position Deviation" feature and related ACTION No. 9.	Operability and related ACTION requirements relocated to LCDs.

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MATRIX OF RELOCATED TECHNICAL SPECIFICATIONS (TS) REQUIREMENTS AND DETAILED TS REQUIREMENTS

Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.3 LA7	Table 3.5-2 Item 15B. Control Rod Misalignment Monitor function as provided by the "Quadrant Power Tilt Monitor" and related ACTION No. 10.	Operability and related ACTION requirements relocated to LCDs.
ITS 3.3 R1	3.8.1.c "During refueling operations"..."radiation levels in the containment and the spent fuel storage areas shall be monitored continuously."	LCDs
ITS 3.3 R1	Table 3.5-5 Item No. 3 and 12, Notes 2 and 7	LCDs
ITS 3.3 R1	Table 3.5-5 Item No. 7 and Note 4	Offsite Dose Calculation Manual (ODCM)
ITS 3.3 R1	Table 4.1-1 Item No. 38	ODCM
ITS 3.3 R1	Table 4.1-1 Item No. 34 and 48 and associated testing requirements	LCDs
ITS 3.4 LA1	3.1.2.1.a "Over the temperature range from cold shutdown to hot operating conditions, the heatup rate shall not exceed 60°F in any one hour."	LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.4 LA1	3.1.2.1.b "Allowable combinations of pressure and temperature for a specific cooldown rate are below and to the right of the limit lines for that rate as shown on Figure 3.1-2. This rate shall not exceed 100°F/hr in any one hour. The limit lines for cooling rates between those shown in Figure 3.1-2 may be obtained by interpolation."	LCDs
ITS 3.4 LA1	3.1.2.1.c "Primary system hydrostatic leak tests may be performed as necessary, provided the temperature limitation as noted on Figure 3.1-1 is not violated. Maximum hydrostatic test pressure should remain below 2350 psia."	LCDs
ITS 3.4 LA1	3.1.2.4.a Requirements for maintaining the pressure-temperature limit curves in TS Figures 3.1-1 and 3.1-2	LCDs
ITS 3.4 LA1	3.1.2.4.b Reporting requirements of results of irradiated specimen samples analysis and updated heatup and cooldown curves	Plant Administrative Procedures for Reporting Information to the NRC

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.4 LA2	3.1.1.1.a.1 The specific 4% shutdown margin when < 2% Rated Thermal Power (RTP) and < 2 reactor coolant pumps operating	COLR
ITS 3.4 LA3	3.1.1.3.a "At least one [Pressurizer] Pzr code safety valve shall be operable whenever the Reactor Head is on the vessel and the [Reactor Coolant System] RCS is not open for maintenance."	LCDs
ITS 3.4 LA4	4.2.4.1.a Requirement that each Pressurizer Power Operated Relief Valve (PORV) be demonstrated operable by performing a channel calibration at each refueling	LCDs
ITS 3.4 LA4	4.2.4.3 Requirement to demonstrate that the nitrogen accumulators for the Pressurizer PORVs are operable by cycling the PORVs through one complete cycle at each refueling	LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.4 LA5	3.1.5.4.a Requirement that all pressure isolation valves listed in Table 3.1-1 be functional as a pressure isolation device except as specified in TS Section 3.1.5.4.b during reactor operation and hot shutdown conditions	LCDs
ITS 3.4 LA5	3.1.5.4.b The compensatory measure for a non-functional pressure isolation valve; "Manual valves shall be locked in the closed position."	LCDs
ITS 3.4 LA5	Table 4.1-3 Item No. 17 Requirement to perform Primary Coolant System check valve tests after maintenance, repair or replacement work is performed	LCDs
ITS 3.4 LA5	Table 4.1-3 Item No. 17.1 Note a. Allowance that Pressure Isolation Valve (PIV) leakage may be measured indirectly if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve leakage compliance	LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.4 LA5	Table 4.1-3 Item No. 17.1 Note b. Minimum test differential pressure for PIVs shall be ≥150 psid.	LCDs
ITS 3.4 LA5	Table 4.1-3 Item No. 17.1 Note c. Allowance that more than one PIV may be tested in parallel provided the total leakage does not exceed 5 gpm	LCDs
ITS 3.4 LA5	Table 4.1-3 Item No. 17.2 Requirement to test the redundant valve in any line containing a PIV which does not meet leakage criteria daily. In addition the position of the redundant valve shall be recorded daily.	LCDs
ITS 3.4 LA5	Table 3.1-1 Listing of safety injection system PIVs	LCDs
ITS 3.4 LA6	Table 4.1-2 Item No. 1 Requirement to perform reactor coolant radiochemical test on a monthly frequency	LCDs
ITS 3.4 LA6	Table 4.1-2 Item No. 2 Requirement to perform reactor coolant boron concentration test on a twice/week frequency	LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.4 LA6	Table 4.1-2 Note (1) Description of a gross activity analysis.	LCDs
ITS 3.4 LA6	Table 4.1-2 Note (2) Description of a radiochemical analysis	LCDs
ITS 3.4 LA7	Table 4.1-2 Item No. 4 Requirement to perform Boric Acid Tank boron concentration test on a twice/week frequency	LCDs
ITS 3.4 LA8	Table 4.1-3 Item No. 14 Requirement to perform fan functional test and laboratory tests of filter media on a once per operating cycle frequency	LCDs
ITS 3.4 R1	Table 4.1-2 Item No. 1 Requirement to sample the reactor coolant system for Chloride and Oxygen on a frequency of 5 times per week	LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.4 R1	Table 4.1-2 Item No. 9 and Note No. 3 Requirement to sample Stack Gas Iodine and Particulate on a weekly bases when iodine or particulate radioactivity levels exceed 10% of the limit in TS Section 3.9.2.1, the sampling frequency shall be increased to a minimum of once per day	LCDs
ITS 3.5 LA1	3.3.1.2.e Specific exclusion of the safety injection hot injection pathways and valves from the requirements of TS Section 3.3.1.2	ITS Section 3.5.2 Bases
ITS 3.6 LA1	1.7.a Details specifying non-automatic isolation valves be closed and blind flanges be properly installed	UFSAR
ITS 3.6 LA2	4.5.1.3 "The test shall be performed with the isolation valves in the spray supply lines at the containment and spray additive tank blocked closed."	LCDs
ITS 3.6 LA3	1.7.d "Manual valves qualifying as automatic containment isolation valves are secured closed."	LCDs
ITS 3.6 LA4	3.6.4.3 Details of testing of 42 inch purge valves	LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.6 R1	3.3.5 "The reactor shall not be made critical unless the valves of the post accident containment venting system are operable."	LCDs
ITS 3.7 LA1	Table 4.1-3 Item No. 12, requirement to check closure of Turbine Steam Stop, Control, Reheat Stop, and Interceptor Valves on a quarterly frequency	LCDs
ITS 3.7 LA2	3.13.1.a Requirement that when the reactor is at power or hot shutdown, if a snubber is determined to be inoperable and an engineering evaluation cannot validate the operability of the supported component then the supported component shall be declared inoperable. If operability can be validated then the snubber shall be returned to operable status within 72 hours.	Inservice Inspection (ISI) Program

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.7 LA2	3.13.1.b "If a snubber is determined to be inoperable while the reactor is in cold shutdown, the snubber (if needed for a supported component protection) shall be repaired and reinstalled or replaced prior to reactor startup."	ISI Program
ITS 3.7 LA3	4.15.a Requirements to verify Control Room air temperature every 12 hrs.	LCDs
ITS 3.7 LA4	5.4.3 "This minimum boron concentration ensures subcriticality under worst case design events," and references	UFSAR
ITS 3.7 LA5	3.12 Seismic shutdown	LCDs
ITS 3.8 LA1	3.7.1.d Description of specific automatic trips that are required to be bypassed for an operable Emergency Diesel Generator (EDG).	LCDs
ITS 3.8 LA1	3.7.1.e and 4.6.1.3 Description of specific automatic trips that are required to be bypassed for an operable EDG. (As referenced to CTS 3.7.1.d)	LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.8 LA2	3.7.3 Restriction allowing the back-feeding of the emergency busses via the unit auxiliary transformer only while the reactor is in cold shutdown unless nuclear safety considerations require it to be done during hot shutdown	LCDs
ITS 3.8 LA3	4.6.1.3 Details describing how the testing of the EDG automatic trips "trips defeat" feature is accomplished	LCDs
ITS 3.8 LA4	4.6.1.3 "Each diesel generator shall be inspected a least once every refueling interval."	LCDs
ITS 3.8 LA5	4.6.1.4.a Details describing the continuous load limits of the EDGs and restriction preventing the continuous operation of the EDGs above these limits	LCDs
ITS 3.8 LA5	4.6.1.4.b Details describing the short-term load limitations of the EDGs and restriction preventing the operation above this limit	LCDs
ITS 3.8 LA5	4.6.1.5 Details describing how the EDG is started and synchronized to the bus in preparation for the 24 hour full load test	LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.8 LA6	Table 4.1-2 Items No. 11 and 12 Surveillance Requirements (SR) and Frequencies for EDG fuel oil testing.	The Diesel Fuel Oil Testing Program
ITS 3.8 LA7	4.6.3.2 Detail describing the precision at which the cell voltage must be determined and the requirement that the amount of water added to each cell be measured and recorded.	LCDs
ITS 3.8 LA7	4.6.3.4 Requirement that when battery data is recorded, the new data shall be compared to previous data in order to detect signs of abuse or deterioration.	LCDs
ITS 3.8 LA8	3.7.1.a and 3.7.1.c Regarding 110 KU-4160 V startup transformer in service and 4160 V buses 2 and 3 energized.	LCDs
ITS 3.8 LA9	3.7.1.b Regarding 480 V buses E1 and 2 energized	LCDs
ITS 3.8 LA10	3.7.1.e Regarding details of the batteries and battery charger	LCDs
ITS 3.9 LA1	3.8.1.f Reference to specific boron concentration of 1950 ppm to be maintained during head removal or movement of fuel in the reactor	COLR

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.9 LA2	Table 4.1-3 Item No. 6 Requirement to test refueling system interlocks prior to each refueling system shutdown	LCDs
ITS 3.9 LA3	3.10.8.3 "When the reactor is in the cold shutdown condition, the shutdown margin shall be a least 1 percent $\Delta k/k$."	COLR
ITS 3.9 LA4	3.8.1.d Requirement that whenever core geometry is being changed that the source range channels provide "continuous visual indication in the control room and one with audible indication available in the containment"	ITS Bases Section 3.9.2.
ITS 3.9 LA5	3.8.1.i Requirement that during refueling operations, when the containment purge system is in operation, the system shall discharge through High Efficiency Particulate Air (HEPA) and impregnated charcoal filters	LCDs
ITS 3.9 LA6	3.8.1.e Requirement that during refueling operations, T_{ave} shall be maintained $\leq 140^{o}$ F	LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 3.9 LA7	3.8.1.h "Movement of fuel within the core shall not be initiated prior to 100 hours after shutdown."	LCDs
ITS 3.9 LA8	3.8.2.c.2 At least one containment purge filter fan ". . . must be operable during core alterations or movement of irradiated fuel assemblies."	LCDs
ITS 3.9 R1	3.8.1.c Requirement that during refueling operations "Radiation levels in the containment and spent fuel storage areas shall be monitored continuously."	LCDs
ITS 3.9 R1	3.8.1.g Requirement that whenever core geometry is being changed, direct communications between the control room and the refueling cavity manipulator crane shall be available	LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 4.0 LA1	5.1 Specifics describing the location of H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2 in relation to HBRSEP Unit No. 1 and the fact that Unit No. 2 is owned and operated by Carolina Power & Light Co. In addition the statement describing site exclusion boundary (per 10 CFR 100.3) as being a circle of 1400 ft radius from the reactor center line.	Updated Final Safety Analysis Report (UFSAR)
ITS 4.0 LA2	5.3.1.1 Specifics describing the reactor core i.e. "approximately 68 metric tons", fuel rods "which are pre pressurized," and fuel assemblies each contain "204 fuel rod locations occupied by rods consisting of natural or slightly enriched uranium pellets, solid inert materials, or a combination of the aforementioned"	UFSAR
ITS 4.0 LA2	5.3.1.3 Descriptive details of reload fuel	UFSAR
ITS 4.0 LA2	5.3.2.1 Design code requirements of the RCS	UFSAR

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 4.0 LA2	5.3.2.2 Descriptive information concerning the piping and components of the RCS meet American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV) Class I requirements	UFSAR
ITS 4.0 LA2	5.3.2.3 Descriptive information concerning the nominal volume of coolant contained in the RCS at rated operating conditions	UFSAR
ITS 4.0 LA3	5.4.1 "The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as ASME B&PV Code Class I structures. The spent fuel pit has a stainless steel liner to ensure against loss of water."	UFSAR

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 4.0 LA3	5.4.2.1 Details describing features of the new fuel storage facilities that maintain $K_{eff} < 0.95$ assuming the racks are flooded with pure water i.e., "...additional separation is maintained by use of the storage rack secured location restrictions"..."in order to establish a geometry which ensures that..."	UFSAR
ITS 4.0 LA3	5.4.2.2 Details describing features that maintain K_{eff} less than 0.95 in the spent fuel pool i.e., "a combination of nominal assembly spacing, neutron absorber material between the assemblies, and restrictions on fuel design, integral burnable absorber content, reconstitution, and storage is required to assure that..." and "fuel assemblies with maximum planar enrichments greater than 4.55+0.05 (4.55 nominal) weight percent U_{235} have requirements for minimum integral burnable absorber content"	UFSAR

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 4.0 LA4	5.2.1.1 Descriptive information concerning the purposes of the containment building	UFSAR
ITS 4.0 LA4	5.2.1.2 Descriptive information concerning the design pressure ratings of the containment building	UFSAR
ITS 4.0 LA4	5.2.2.1 Descriptive information concerning the design of the containment penetrations for electrical and mechanical systems	UFSAR
ITS 4.0 LA4	5.2.2.2 Descriptive information concerning the Phase A and Phase B containment isolation signals and the fact that they must be capable of withstanding a single component failure and still maintain containment isolation	UFSAR
ITS 4.0 LA4	5.2.3.1 Descriptive information concerning the containment spray system and its purpose	UFSAR
ITS 4.0 LA4	5.2.3.2 Descriptive information concerning the containment internal air recirculation system and its heat removal capability	UFSAR

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 4.0 LA5	5.5 Descriptive information concerning the design of the containment building, auxiliary building, ASME B&PV Code Class I turbine bay, and all contained Engineered Safety Feature (ESF) systems be designed for a maximum credible earthquake with an acceleration of 0.20 g	UFSAR
ITS 5.0 LA1	6.5.1.1.2 Requirements concerning how safety analysis shall be prepared for all procedures, tests, and experiments covering procedures identified in TS Section 6.5.1.1.1 and procedures that affect nuclear safety	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.1.3 Requirements concerning when a second safety review shall be performed on procedures affecting nuclear safety	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.1.4 Requirements for approval of procedures that do not involve an unreviewed safety question as defined in 10 CFR 50.59 nor a change to the TS	UFSAR and LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 5.0 LA1	6.5.1.1.5 Requirements concerning approval of temporary changes to procedures and the maximum time it may be in effect i.e., 21 days	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.1.6 Requirements concerning changes to procedures that constitute an unreviewed safety question, or involve a change to the TS	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.1.7 Requirements concerning changes which constitute a change to the facility as described in the UFSAR	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.2.1 Requirements concerning plant modifications that affect nuclear safety	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.2.2 Requirement concerning the second safety review on all modifications that affect nuclear safety	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.2.3 Requirements concerning approval of modifications that do not involve an unreviewed safety question or a change to the TS	UFSAR and LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 5.0 LA1	6.5.1.2.4 Requirements concerning approval of modifications that either constitute an unreviewed safety question or a change to the TS	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.3.1 "Each proposed Technical Specification or Operating License change shall be reviewed by the Plant Nuclear Safety Committee and submitted to the NRC for approval."	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.4.1 Requirements concerning Technical Specification violations	UFSAR and LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 5.0 LA1	6.5.1.5.1 Qualification requirements for Nuclear Safety Reviewers	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.6.1.a and b Requirements for establishing a Plant Nuclear Safety Committee (PNSC) and the advisory role it plays to the Plant General Manager	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.6.2 Requirement for the composition of the PNSC	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.6.3 Requirements for PNSC members and alternate members	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.6.4.a and b Definition of "quorum" as it pertains to the PNSC. Also limits the number of alternates that may compose a quorum	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.6.5 Minimum PNSC meeting requirements	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.6.6.a-k Listing of activities requiring PNSC involvement	UFSAR and LCDs
ITS 5.0 LA1	6.5.1.6.7 Required actions to be taken upon disagreement between the PNSC and actions contemplated by the Plant General Manager	UFSAR and LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 5.0 LA1	6.5.1.6.8 Requirement for maintaining minutes of PNSC meetings and their minimum content	UFSAR and LCDs
ITS 5.0 LA1	6.5.2.1 Description of the function of the Nuclear Assessment Section (NAS)	UFSAR
ITS 5.0 LA1	6.5.2.2.1 Qualifications of individuals for independent reviews in the NAS	UFSAR
ITS 5.0 LA1	6.5.2.2.2 Qualifications of the Manager-Nuclear Assessment Section	UFSAR
ITS 5.0 LA1	6.5.2.2.3 Qualifications of individuals performing independent safety reviews	UFSAR
ITS 5.0 LA1	6.5.2.2.4 Actions to be taken when sufficient expertise does not exist within the NAS. Also allows an individual to be "competent" in more than one specialty	UFSAR and LCDs
ITS 5.0 LA1	6.5.2.2.5 Qualifications for individuals performing reviews of documents submitted under TS Section 6.5.2.3	UFSAR
ITS 5.0 LA1	6.5.2.2.6 Requirements for independent safety reviews in NAS	UFSAR and LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 5.0 LA1	6.5.2.2.7 Requirement that the NAS Independent Safety Review Program be conducted in accordance with written, approved procedures	UFSAR and LCDs
ITS 5.0 LA1	6.5.2.3 a-e Listing of items in which the NAS shall perform reviews	UFSAR and LCDs
ITS 5.0 LA1	6.5.2.4 "Results of Nuclear Assessment Section independent safety reviews shall be documented and retained."	UFSAR and LCDs
ITS 5.0 LA2	6.2.1.e Requirement that the health physics manager have access to the overall unit manager and the health physics technician's "stop work" authority	UFSAR and LCDs
ITS 5.0 LA3	6.2.3.h Restriction that minimum shift manning requirements cannot be used to justify adequate shift complement upon shift relief when a required member is not available for relief	UFSAR and LCDs
ITS 5.0 LA4	6.5.1.1.1.j Reference to Regulatory Guide 4.15, Dec. 1977 with regard to the Quality Assurance Program	UFSAR

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 5.0 LA5	Facility Operating License No. DPR-23, paragraph 3.G(3) Requirement to have a program to determine the airborne iodine concentration in vital areas under accident conditions	LCDs
ITS 5.0 LA6	4.4.3.a-h Requirement to perform leakage testing and inspections of the Post Accident Recirculation Heat Removal System and the applicable acceptance criteria. Additional requirement to perform repairs as necessary to maintain leakage within the stated criteria	Primary Coolant Sources Outside Containment Program
ITS 5.0 LA7	4.4.4.1 Requirement to inspect containment surveillance tendons	Prestressed Concrete Containment Tendon Surveillance Program
ITS 5.0 LA7	4.4.4.3.a Details describing analysis to be performed on Containment Surveillance Tendons	Prestressed Concrete Containment Tendon Surveillance Program
ITS 5.0 LA8	4.2.3 Requirement to perform reactor coolant pump flywheel inspections	Reactor Coolant Pump Flywheel Inspection Program

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ITS 5.0 LA9	3.8.2.c Requirement that the spent fuel building and containment building filter fans shall be shown to operate within ±10% of design flow	Ventilation Filter Testing Program
ITS 5.0 LA9	4.15.d Requirement that the control room filtration system be tested following any structural maintenance on the filter housings or following painting, fire, or chemical release in the Control Room envelope	Ventilation Filter Testing Program
ITS 5.0 LA9	4.15.f Requirement that the Control Room filtration system be tested every 18 months	Ventilation Filter Testing Program
ITS 5.0 LA9	4.15.g Requirement that the Control Room filtration system HEPA filters be tested after complete or partial replacement and associated test conditions	Ventilation Filter Testing Program
ITS 5.0 LA9	4.15.h Requirement that the Control Room filtration system charcoal filters be tested after each partial or complete replacement and associated test conditions	Ventilation Filter Testing Program
ITS 5.0 LA9	4.12 Refueling filter systems Applicability and Objective	Ventilation Filter Testing Program

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 5.0 LA9	4.12.1 Requirement that the Refueling Filter System be demonstrated operable every operating cycle	Ventilation Filter Testing Program
ITS 5.0 LA9	4.12.2.a Requirement that the Refueling Filter System be tested initially, and at least once per operating cycle prior to each refueling outage or after 720 hours of system operation whichever comes first	Ventilation Filter Testing Program
ITS 5.0 LA9	4.12.2.b-e Details concerning how the Refueling System filtration system is to be tested and what tests are to be performed	Ventilation Filter Testing Program
ITS 5.0 LA10	3.16.2.1 Restriction on the quantity of radioactive material contained in the listed tanks shall be limited to ≤10 curies excluding tritium and dissolved or entrained noble gases at all times	Explosive Gas and Storage Tank Radioactivity Monitoring Program

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 5.0 LA10	3.16.2.2 Requirement that when the quantity of radioactive material in any listed tank exceeds the limit, suspend all additions to the tank, and take actions to reduce the amount of radioactive material in the tank to within limits within 48 hours	Explosive Gas and Storage Tank Radioactivity Monitoring Program
ITS 5.0 LA10	3.16.2.3 Requirement that if the radioactive content of any of the listed tanks cannot be reduced to within limits within 48 hours then the NRC shall be notified in accordance with TS Section 6.6	Explosive Gas and Storage Tank Radioactivity Monitoring Program
ITS 5.0 LA10	Definition of "Temporary Tank" as it applies to TS Section 3.16.2.1.f	Explosive Gas and Storage Tank Radioactivity Monitoring Program
ITS 5.0 LA10	3.16.4.1 Restrictions on hydrogen and oxygen content in the Waste Gas Decay Tanks	Explosive Gas and Storage Tank Radioactivity Monitoring Program
ITS 5.0 LA10	3.16.4.1.a Actions to be taken when oxygen or hydrogen concentration in the Waste Gas Decay Tanks exceeds limits	Explosive Gas and Storage Tank Radioactivity Monitoring Program

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 5.0 LA10	3.16.4.1.b Actions to be taken when oxygen and hydrogen concentration in the Waste Gas Decay Tanks exceeds limits	Explosive Gas and Storage Tank Radioactivity Monitoring Program
ITS 5.0 LA10	3.16.4.2 Reporting requirements for oxygen and/or hydrogen being out of specification in the Waste Gas Decay Tank(s) for ≥48 hours	Plant Administrative Procedures for Reporting Information to the NRC
ITS 5.0 LA10	3.16.4.3 Reporting requirements for condition where actions required to be taken by TS do not result in returning the hydrogen or oxygen concentration in the Waste Gas Decay Tank(s) to ≤6% within 24 hours	Plant Administrative Procedures for Reporting Information to the NRC
ITS 5.0 LA10	3.16.5.1 Limitation on the amount of radioactive material that may be stored in any one Waste Gas Decay Tank shall be limited to ≤1.9 E+4 curies noble gas (considered as Xe-133)	Explosive Gas and Storage Tank Radioactivity Monitoring Program
ITS 5.0 LA10	3.16.5.2 Requirement to suspend all additions to any Waste Gas Decay Tank that exceeds the limit in TS Section 3.16.5.1 and within 48 hours reduce the tank contents to within limits	Explosive Gas and Storage Tank Radioactivity Monitoring Program

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 5.0 LA10	3.16.5.3 Reporting requirements if the Waste Gas Decay Tank contents are not reduced to within limits within the time period allowed by TS Section 3.16.5.2	Plant Administrative Procedures for Reporting Information to the NRC
ITS 5.0 LA10	4.20.2 Note specifying which tanks are to be included in limitations dictated by TS Section 4.20.2.1	Explosive Gas and Storage Tank Radioactivity Monitoring Program
ITS 5.0 LA10	4.20.2.1 Requirement to verify the radioactive material content of tanks listed in TS Section 3.16.2.1 by sampling	Explosive Gas and Storage Tank Radioactivity Monitoring Program
ITS 5.0 LA10	4.20.4.1 Requirement to verify the hydrogen and oxygen concentration in the Waste Gas Decay Tanks to be within limits by monitoring contents with hydrogen and oxygen monitors or sampling	Explosive Gas and Storage Tank Radioactivity Monitoring Program
ITS 5.0 LA10	4.20.5.1 Requirement to sample the contents of the Waste Gas Decay Tanks every 24 hours when Reactor Coolant Activity is ≥ 100 uCi/ml	Explosive Gas and Storage Tank Radioactivity Monitoring Program
ITS 5.0 LA11	not used	not used

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 5.0 LA12	6.5.3.1 Requirement to perform certain types of assessments at a frequency not to exceed 24 months, by the NAS as listed in TS Sections 6.5.3.1 a-h	UFSAR and LCDs
ITS 5.0 LA12	6.5.3.2 Requirement for the NAS to perform assessments in accordance with the Code of Federal Regulations specifically in the areas of Emergency Preparedness, Security, and Radiation Protection	UFSAR and LCDs
ITS 5.0 LA12	6.5.4.1 Requirement for an independent fire protection and loss prevention inspection and audit to be performed at least every 12 months by qualified offsite personnel	UFSAR and LCDs
ITS 5.0 LA13	6.9.1.1 Reporting requirements for plant startup and power escalation	Plant Administrative Procedures for Reporting Information to the NRC
ITS 5.0 LA14	6.9.1.3.7 Reporting and approval requirements for changes to the radioactive waste systems	Plant Administrative Procedures for Reporting Information to the NRC

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 5.0 LA14	6.17.1.1 Reporting requirements for major changes to the radioactive liquid, gaseous, or solid waste treatment systems. Includes allowance that licensee may submit the required information as part of the next UFSAR Update	Plant Administrative Procedures for Reporting Information to the NRC
ITS 5.0 LA14	6.17.1.2 Major changes to radioactive waste process systems shall become effective upon review and approval by the PNSC	LCDs
ITS 5.0 LA15	6.9.3.3.b Listing of TS requirements applicable to each listed methodology as they pertain to determining core operating limits	COLR
ITS 5.0 LA16	6.9.3.2 Requirement to generate written reports for the listed special radiological effluent reports listed and submit to the NRC within 30 days of the occurrence or event	Plant Administrative Procedures for Reporting Information to the NRC
ITS 5.0 LA17	1.15 Requirements for the Process Control Program (PCP) to assure compliance with 10 CFR 20, 10 CFR 17, and Federal and State regulations	PCP

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 5.0 LA17	6.15.1 Requirement that the PCP shall be approved by the NRC prior to implementation	PCP
ITS 5.0 LA17	6.15.2 Reporting and approval requirements for changes to the PCP	Plant Administrative Procedures for Reporting Information to the NRC
ITS 5.0 LA18	6.4.1 Requirements for a requalification and replacement training program for the plant staff that shall meet the requirements of Sec. 5.5 of ANSI N18.1-1971 and 10 CFR 55, Appendix A	UFSAR and LCDs
ITS 5.0 LA19	6.10.1.a-i Listing of facility records that must be retained for at least five years	UFSAR and LCDs
ITS 5.0 LA19	6.10.2.a-m Listing of facility records that must be retained for the duration of the facility operating license	UFSAR and LCDs
ITS 5.0 LA20	6.11 Requirement that procedures for personnel radiation protection for the Radiation Protection Program be prepared consistent with the requirements of 10 CFR 20 and that they be approved, maintained, and adhered to for all operations involving personnel radiation exposure	UFSAR and LCDs

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Improved Technical Specifications (ITS) Relocation No.	Current Technical Specification (CTS) Section Requirement	Document(s) to which the requirement was relocated
ITS 5.0 LA21	6.1.1, 6.5.1.1.4, 6.5.1.2.3, 6.2.1(b), 6.2.3, 6.3.1, 6.16.2, 6.13.1, and 6.13.2, With respect to detailed plant organization titles	LCDs
ITS 5.0 LA 22	6.9.1.2, 6.9.1.2.3, and 6.9.1.3, With regard to reporting requirements for radiological effluents	Plant Administrative Procedures for Reporting Information to the NRC
ITS 5.0 LA 23	6.9.3.3.a With regard to mathematical terms utilized in the COLR	COLR

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RELOCATED ITEMS WITHOUT CORRESPONDING ITS

CTS Requirements	Relocated To the Following Document(s)
3.1.1.4.A Requirement that when the RCS temperature is >200°F RCS vent paths from the reactor vessel head and pressurizer steam space shall be operable	LCDs
3.1.1.4.B Requirement that when the RCS temperature is >200°F, valves RC-571 and 572 shall be closed with the allowance that the valves may be cycled periodically in order to depressurize the vent system should the system pressurize due to "root" valve leak-by	LCDs
3.1.1.4.C.1 "With the Reactor Vessel Head vent path inoperable, restore the vent path to operable status within 30 days or be in Hot Shutdown within 6 hours and Cold Shutdown within the following 30 hours."	LCDs
3.1.1.4.C.2 Reporting requirement that with the Pressurizer steam space vent inoperable, restore it to operable status within 30 days or prepare and submit a special report to the NRC within the following 14 days detailing the cause of the inoperability and the action being taken to restore operability	LCDs
3.1.6.1 "The concentration of oxygen in the reactor coolant shall not exceed 0.1 ppm when the reactor coolant temperature exceeds 250°F."	LCDs
3.1.6.2 "The concentration of chloride in the reactor coolant shall not exceed 0.15 ppm when the reactor coolant temperature exceeds 250°F."	LCDs

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CTS Requirements	Relocated To the Following Document(s)
3.1.6.3 Requirement that if out of specification reactor coolant oxygen or chloride concentration cannot be restored to within specifications within 24 hours the unit shall be placed in the cold shutdown condition using normal operating procedures	LCDs
3.1.2.2 "The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 120°F."	LCDs
3.1.2.3 "The pressurizer shall neither exceed a maximum heatup rate of 100°F/hr nor a cooldown rate of 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F."	LCDs
3.5.2.1 Requirement that the equipment listed in TS Table 3.5-6 shall be operable with their alarm/trip setpoints set in accordance with the ODCM	ODCM
3.5.2.2 Requirement that with the channel setpoint less conservative than that required in TS Section 3.5.2.1, immediately suspend releases via the associated pathway and restore the channel setpoint to that required by the ODCM or declare the channel inoperable	ODCM
3.5.2.3 "With less than the minimum number of radioactive liquid effluent monitoring instrumentation operable, take the action shown in Table 3.5-6."	ODCM
3.5.2.4 "The provisions of the Specification 3.0 are not applicable."	ODCM

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CTS Requirements	Relocated To the Following Document(s)
3.5.3.1 Requirement that the equipment listed in TS Table 3.5-7 shall be operable with their alarm/trip setpoints set in accordance with the ODCM	ODCM
3.5.3.2 Requirement that with the channel setpoint less conservative than that required in TS Section 3.5.3.1, immediately suspend releases via the associated pathway and restore the channel setpoint to that required by the ODCM or declare the channel inoperable	ODCM
3.5.3.3 "With less than the minimum number of radioactive gaseous effluent monitoring instrumentation operable, take the action shown in Table 3.5-6."	ODCM
3.5.3.4 "The provisions of the Specification 3.0 are not applicable."	ODCM
Table 3.5-6 "Radioactive Liquid Effluent Monitoring Instrumentation," which lists those monitors required to monitor the various liquid radioactive release pathways and the required actions to be taken when the monitoring channel is inoperable. The table also provides the necessary compensatory action that must be taken if it is desired to maintain the release via the pathway with the associated monitor inoperable.	ODCM

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CTS Requirements	Relocated To the Following Document(s)
Table 3.5-7 "Radioactive Gaseous Effluent Monitoring Instrumentation," which lists those monitors required to monitor the various gaseous radioactive release pathways and the required actions to be taken when the monitoring channel is inoperable. The table also provides the necessary compensatory action that must be taken if it is desired to maintain the release via the pathway with the associated monitor inoperable.	ODCM
3.9.1.1 Requirement that the concentration of radioactive material in liquid effluents released at any time from the site to unrestricted areas shall be limited to the concentrations specified in 10 CFR 20, App. B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to $2x10^{-4}$ uCi/ml total activity.	LCDs
3.9.1.2 Requirement the with the concentration of liquid effluents in excess of that allowed by TS Section 3.9.1.1, without delay restore the concentration to within limits and notify the NRC in accordance with TS Section 6.6.	LCDs
3.9.1.3 Requirement that in the event the requirements of TS Section 3.9.1.2 cannot be met the unit shall be placed in the hot shutdown condition.	LCDs
3.9.1.4 "The provisions of Specification 3.0 are not applicable."	Radioactive Effluent Controls Program

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CTS Requirements	Relocated To the Following Document(s)
3.9.2.1 Requirement that the dose commitment at all times to a member of the public from radioactive materials in liquid effluents released to unrestricted areas shall be limited as stated in (a) and (b)	Radioactive Effluent Controls Program
3.9.2.2 "With the calculated dose commitment from the release of radioactive materials in liquid effluents exceeding any of the limits prescribed by Specification 3.9.2.1 above, prepare and submit a report to the Commission in accordance with Specification 6.9.3.2."	Radioactive Effluent Controls Program and Plant Administrative Procedures for Reporting information to the NRC
3.9.3.1 Requirement that the dose rate at all times to a member of the public from radioactive materials in gaseous effluents released from the site boundary shall be limited as stated in (a) and (b)	Radioactive Effluent Controls Program
3.9.3.2 "With the dose rate(s) exceeding the above limits, without delay decrease the release rate to within the above limits. In addition, a notification must be made to the Commission in accordance with Specification 6.6."	Radioactive Effluent Controls Program and Plant Administrative Procedures for Reporting information to the NRC
3.9.3.3 Requirement that in the event the requirements of TS Section 3.9.3.2 cannot be met the unit shall be placed in the hot shutdown condition	Radioactive Effluent Controls Program and TRM
3.9.4.1 Requirement that the air dose commitment due to radionoble gases released in gaseous effluents to areas at and beyond the site boundary shall be limited as stated in (a) and (b)	Radioactive Effluent Controls Program

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CTS Requirements	Relocated To the Following Document(s)
3.9.4.2 Requirement that with the limits of TS Section 3.9.4.1 exceeded, prepare and submit a report to the NRC in accordance with TS Section 6.9.3.2	Radioactive Effluent Controls Program and Plant Administrative Procedures for Reporting information to the NRC
3.9.5.1 Requirement that the dose to the public from I-131, I-133, tritium, and radioactive materials in particulate form, with half-lives greater than eight (8) days be limited in accordance with limits stated in (a) and (b)	Radioactive Effluent Controls Program
3.9.5.2 Requirement that with the limits of TS Section 3.9.5.1 exceeded, prepare and submit a report to the NRC in accordance with TS Section 6.9.3.2	Radioactive Effluent Controls Program and Plant Administrative Procedures for Reporting information to the NRC
3.9.6.1 Requirement that the dose commitment to any member of the public, due to releases of licensed materials and radiation, from uranium fuel cycle sources shall be limited to ≤25 mrem to the total body or any organ except the thyroid, which shall be limited to ≤75 mrem over 12 consecutive months	Radioactive Effluent Controls Program
3.9.6.2 Reporting and analysis requirements when any of the limits stated in TS Sections 3.9.2.1.a or b,3.9.4.1.a or b, or 3.9.5.1.a or b are exceeded by a factor of 2	Radioactive Effluent Controls Program and Plant Administrative Procedures for Reporting information to the NRC
3.9.6.3 "The provisions of Specification 3.0 are not applicable."	Radioactive Effluent Controls Program
3.11.1 "A minimum of 15 total accessible thimbles and at least 2 per quadrant sufficient movable in-core detectors shall be operable during recalibration of the excore symmetrical offset detection system."	LCDs

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MATRIX OF RELOCATED TECHNICAL SPECIFICATIONS (TS) REQUIREMENTS AND DETAILED TS REQUIREMENTS

CTS Requirements	Relocated To the Following Document(s)
3.11.2 "Power shall be limited to 90% of rated power if recalibration requirements for the excore symmetrical offset detection system identified in Table 4.1-1 are not met."	LCDs
3.12 Requirement that when the strong motion recorder indicated that the operating basis earthquake has been exceeded, the reactor shall be shut down and shall remain shut down until inspection of the facility shows that no damage has been incurred which would jeopardize safe operation of the facility or until such damage is repaired	LCDs
3.16.1.1 Requirement that the radioactive liquid waste system be utilized to minimize offsite doses due to releases of liquid radioactive effluents	Radioactive Effluent Controls Program
3.16.1.2 Requirement that with liquid wastes being discharged without treatment while in excess of the limits in TS Section 3.16.1.1, prepare and submit a report to the NRC in accordance with TS Section 6.9.3.2.b	Radioactive Effluent Controls Program
3.16.3.1 Requirement that the radioactive gaseous waste system be utilized to minimize offsite doses due to releases of liquid radioactive effluents	Radioactive Effluent Controls Program
3.16.3.2 Requirement that with the gaseous waste treatment system inoperable and gaseous releases in excess of the limits in 3.16.3.1, prepare and submit a special TS Section report to the NRC in accordance with TS Section 6.9.3.2.b	Radioactive Effluent Controls Program and Plant Administrative Procedures for Reporting information to the NRC

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CTS Requirements	Relocated To the Following Document(s)
3.16.6.1 "The Solid Radioactive System shall be used in accordance with a Process Control Program (PCP) to process wet radioactive waste to meet shipping and burial ground requirements."	LCDs
3.16.6.2 "With the provisions of the PCP not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive waste from the site."	LCDs
3.16.6.3 Requirement that if a test specimen fails to verify solidification, the solidification of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative solidification parameters can be determined in accordance with the PCP, and a subsequent test verifies solidification	LCDs
3.17.1.1 "The Radiological Environmental Monitoring Program shall be conducted as specified in Table 3.17-1."	ODCM
3.17.1.2 Actions to be taken when the Radiological Environmental Monitoring Program is not in accordance with TS Section 3.17-1	ODCM
3.17.1.3 Actions to be taken when the level of radioactivity in plant effluents as indicated by environmental sampling is greater than the reporting levels of TS Table 3.17-2	ODCM
3.17.1.4 Requirement to obtain replacement samples of milk and leafy vegetables when the sample locations specified in TS Table 3.17-1 cannot be obtained	ODCM

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MATRIX OF RELOCATED TECHNICAL SPECIFICATIONS (TS) REQUIREMENTS AND DETAILED TS REQUIREMENTS

CTS Requirements	Relocated To the Following Document(s)
3.17.1.5 "The provisions of Specification 3.0 are not applicable."	ODCM
3.17.1.6 Allowance for deviations from the required environmental sampling schedule	ODCM
3.17.2.1 Requirement to perform a land use census and related content	ODCM
3.17.2.2 Reporting requirement for land use census that identifies doses greater than the values currently being calculated in TS Section 4.10.4.1	ODCM
3.17.2.3.a-c Actions to be taken when the land use census identifies a location which yields an annual calculated dose or dose commitment of a specific pathway which is 20% greater than that at a current sampling location	ODCM
3.17.3.1 Requirement for analyses supplied by U. S. Environmental Protection Agency (EPA) as a part of Interlaboratory Comparison Program	ODCM
3.17.3.2 Reporting requirements when the Interlaboratory Comparison Program is not conducted as described in TS Section 3.17.3.1	ODCM
Table 3.17-1 Table and associated notation describing the Radiological Environmental Monitoring Program	ODCM
Table 3.17-2 Table describing reporting levels for radioactivity concentrations in environmental samples	ODCM

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MATRIX OF RELOCATED TECHNICAL SPECIFICATIONS (TS) REQUIREMENTS AND DETAILED TS REQUIREMENTS

CTS Requirements	Relocated To the Following Document(s)
4.10.1.1 Sampling requirements for batch liquid releases	ODCM
4.10.1.2 Analysis requirements for samples of batch liquid releases	ODCM
4.10.1.3 Requirement that liquid batch samples be taken and analyzed in accordance with TS Table 4.10-1. Requirement that the concentrations at the point of release be within the limits of TS Section 3.9.1.1	ODCM
4.10.2.1 Requirement for determining the dose rate due to radioactive materials in gaseous effluents to be within the limits of TS Section 3.9.3.1 by performing sampling and analysis in accordance with Table 4.10-2	ODCM
4.10.3.1 "Cumulative dose commitments for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM once per 31 days."	ODCM
4.10.4.1 Requirement to determine cumulative dose contributions for the current quarter and current calendar year for I-131, I-133, tritium, and radionuclides in particulate form with half lives greater than 8 days	ODCM
4.10.5.1 Requirement to determine cumulative dose contributions from liquid and gaseous effluents in accordance with TS Sections 3.9.2.1, 3.9.4.1, and 3.9.5.1	ODCM

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MATRIX OF RELOCATED TECHNICAL SPECIFICATIONS (TS) REQUIREMENTS AND DETAILED TS REQUIREMENTS

CTS Requirements	Relocated To the Following Document(s)
4.10.5.2 Requirement to determine cumulative dose contributions from direct radiation from the reactor unit and from radwaste storage tanks as set forth in the applicability of TS Section 3.9.6.2	ODCM
Table 4.10-1 Table and associated notation describing the Radioactive Liquid Waste Sampling and Analysis Program	ODCM
Table 4.10-2 Table and associated notation describing the Radioactive Gaseous Waste Sampling and Analysis Program	ODCM
4.19.1.1 Requirement to perform radioactive liquid effluent monitoring instrumentation operability surveillances in accordance with TS Table 4.19-1	ODCM
4.19.2.1 Requirement to perform radioactive gaseous effluent monitoring instrumentation operability surveillances in accordance with TS Table 4.19-2	ODCM
Table 4.19-1 Table and associated notation describing the radioactive liquid effluent monitoring instrumentation surveillance requirements	ODCM
Table 4.19-2 Table and associated notation describing the radioactive gaseous effluent monitoring instrumentation surveillance requirements	ODCM
4.20.1.1 Requirements to perform projected dose commitments at least every 31 days to ensure the requirements of TS Section 3.16.1.1 are satisfied when the Liquid Radwaste Treatment System is not used	ODCM

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MATRIX OF RELOCATED TECHNICAL SPECIFICATIONS (TS) REQUIREMENTS AND DETAILED TS REQUIREMENTS

CTS Requirements	Relocated To the Following Document(s)
4.20.3.1 Requirements to perform projected dose commitments for gaseous releases at least every 31 days to ensure the requirements of TS Section 3.16.3.1 are satisfied	ODCM
4.20.6.1 "The PCP shall be used to verify the solidification of one representative test specimen from every tenth batch of wet radioactive waste."	PCP
4.20.6.2 Actions to be taken when a test specimen from a batch of waste fails to verify solidification	PCP
4.21.1.1 Requirement to collect environmental samples in accordance with TS Table 3.17-1 and analyze them in accordance with TS Tables 3.17-2 and 3.17-3	ODCM
4.21.2.1 Specification of how the land use census shall be conducted	ODCM
4.21.3.1 Requirement to perform analyses as part of the EPA Interlaboratory Comparison Program	ODCM
TS Appendix B "Radioactive Effluent Releases" requirement to report on a monthly basis the quantities of radioactive effluents released from the plant	ODCM

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
RCS OVERPRESSURE PROTECTION ANALYSIS
WITH ONE SAFETY INJECTION (SI) PUMP

H.B. Robinson Steam Electric Plant, Unit No. 2
RCS Overpressure Protection Analysis
with One Safety Injection (SI) Pump

August 1996

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RCS Overpressure Protection Analysis with One Safety Injection (SI) Pump

1.0 INTRODUCTION

This report documents an analysis of the operation of the Low Temperature Overpressure Protection (LTOP) system in the event a single Safety Injection (SI) pump starts concurrent with one operating charging pump. The LTOP system is in service when the primary system temperature is below 350°F and supplements the Residual Heat Removal (RHR) system relief valves and the coolant letdown system to prevent overpressurization of the Reactor Coolant System (RCS). When the LTOP system is placed into service it adjusts the operating setpoints of the Power Operated Relief Valves (PORV) to provide the protection. The existing analysis for H.B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, was based on generic vessel cooldown limits and the more restrictive curves, which resulted in the current Technical Specifications that require the SI pumps be disabled when the LTOP is placed into service. Consequently, current Technical Specifications remove any mechanism for mitigating a Loss of Coolant Accident (LOCA) in Mode 4 automatically.

This analysis summary demonstrates that without taking credit for the operation of the RHR relief valves or the coolant letdown system, and assuming the cooldown limitation curves from the current Technical Specifications (Reference 1), the LTOP system provides protection against the inadvertent operation of a single charging pump and one SI pump.

2.0 SUMMARY AND RESULTS

The cases analyzed were selected to bound the operating conditions allowed for Mode 4 (200°F < T_{ave} ≤ 350°F and P ≤ 400 psig) or Mode 5 (175°F < T_{ave} ≤ 200°F and P ≥ 275* psig) and the range of allowable flows. The maximum flow supported over the entire range corresponds to the operation of two of the Reactor Coolant Pumps (RCP) and the minimum to the minimum RHR flow. No credit was taken for the operation of the RHR relief valves or the coolant letdown system and only one of the two PORVs was assumed to operate.

The cases analyzed are summarized in Table 2.1. The two flow conditions represent maximum and minimum flows for Modes 4 and 5. The flow for two RCPs is the maximum flow because operation of three RCPs would have resulted in negative margins at the lower temperature (175°F) due to the increased dynamic head between the pressurizer and the lower vessel head. The pressurization limits correspond to the 0°F cooldown limitation curve in Figure 3.1-2 of Reference 1.

Table 2.1 LTOP System Analysis Results

Inlet Conditions		Peak Reactor Vessel Lower Head Pressure (psig)	Pressure Limit (psig)	Margin to Limit (psi)
Pressure (psig)	Temperature (°F)			
Cases with Two RCPs Operating				
400	350	535.5	2,030	1,494.5
400	175	560.2	570	9.8
275	350	536.5	2,030	1,493.5
275	175	562.4	570	7.6
Cases with Minimum RHR Flow				
400	350	494.2	2,030	1,535.5
400	175	514.7	570	55.3
275	350	495.0	2,030	1,535.0
275	175	515.0	570	55.0

3.0 ANALYSIS

The analysis used a modified ANF-RELAP model which was based on the HBRSEP, Unit No. 2 Cycle 17 Non-LOCA transient model. This model was constructed in accordance with Siemens Power Corporation's (SPC) methodology (Reference 2) for analyzing events from the Standard Review Plan (SRP)(Reference 3).

3.1 Modeling Changes

Modeling changes were incorporated in the ANF-RELAP deck to reflect the design information and assumptions below.

PORV

The operation of the PORV assumes no dynamic compensation and has opening and closing setpoints of 430 psig and 415 psig, respectively. These values incorporate a 30 psi error allowance on the 400 psig opening setpoint and a hysteresis in the closing pressure of 30 psi $\pm$ 15 psi. A fixed delay of 0.6 seconds was used in modeling the dynamic response after which the PORV cycled to full open in 2.4 seconds. The flow for the PORV was based on the relation,

$$F_{PORV} = (50gpm) x \sqrt{\Delta P} \quad ,$$

where ΔP is the difference between the pressurizer pressure and a 10 psig back-pressure (Reference 4).

SI and Charging Pumps

A fixed flow of 77 gpm was used to model the flow from one charging pump. The nominal SI delivery curves were adjusted by selecting the lowest head for each flow rate to create a composite curve and then increasing the resulting flow by 5%.

Pumps

One RCP was turned off for the first four cases in Table 2.1 and all three were turned off for the last

four cases. The RHR was simulated for these last four cases by removing flow from one hot leg and re-injecting it in the cold leg. This provided minimal flow and mixing of the injected water with the primary coolant.

Pressurizer

The pressurizer was initialized as a water filled volume to maximize the pressure transient. The pressurizer heaters were disabled.

Other

The kinetics model, the pressurizer safety valves and dynamic compensation for the PORV were not used in the analysis.

3.2 Analysis

Each of the eight cases summarized in Table 2.1 were analyzed to determine the maximum pressure in the lower plenum of the reactor vessel. This pressure conservatively bounds the pressure at the reactor vessel beltline. Figure 3.1 shows the pressure trace in the lower plenum of the vessel during the event. Note that the pressure is somewhat higher than the pressurizer pressure of 275 psi. This is due to the elevation difference (about 68 feet) and the dynamic head of the two RCPs. The second pressure peak is a result of the LTOP de-pressurizing until it reaches the PORV closure point. Once the PORV closed, the system pressure began to rise again until the LTOP setpoint was reached and the PORV opened again.

The heatup and cooldown curves in Reference 1 were used to set the pressure limits. These curves contain a 60 psi allowance for instrument error. The actual error is 30 psi and the values extracted from the curves in Reference 1 were adjusted upwards by 30 psi.

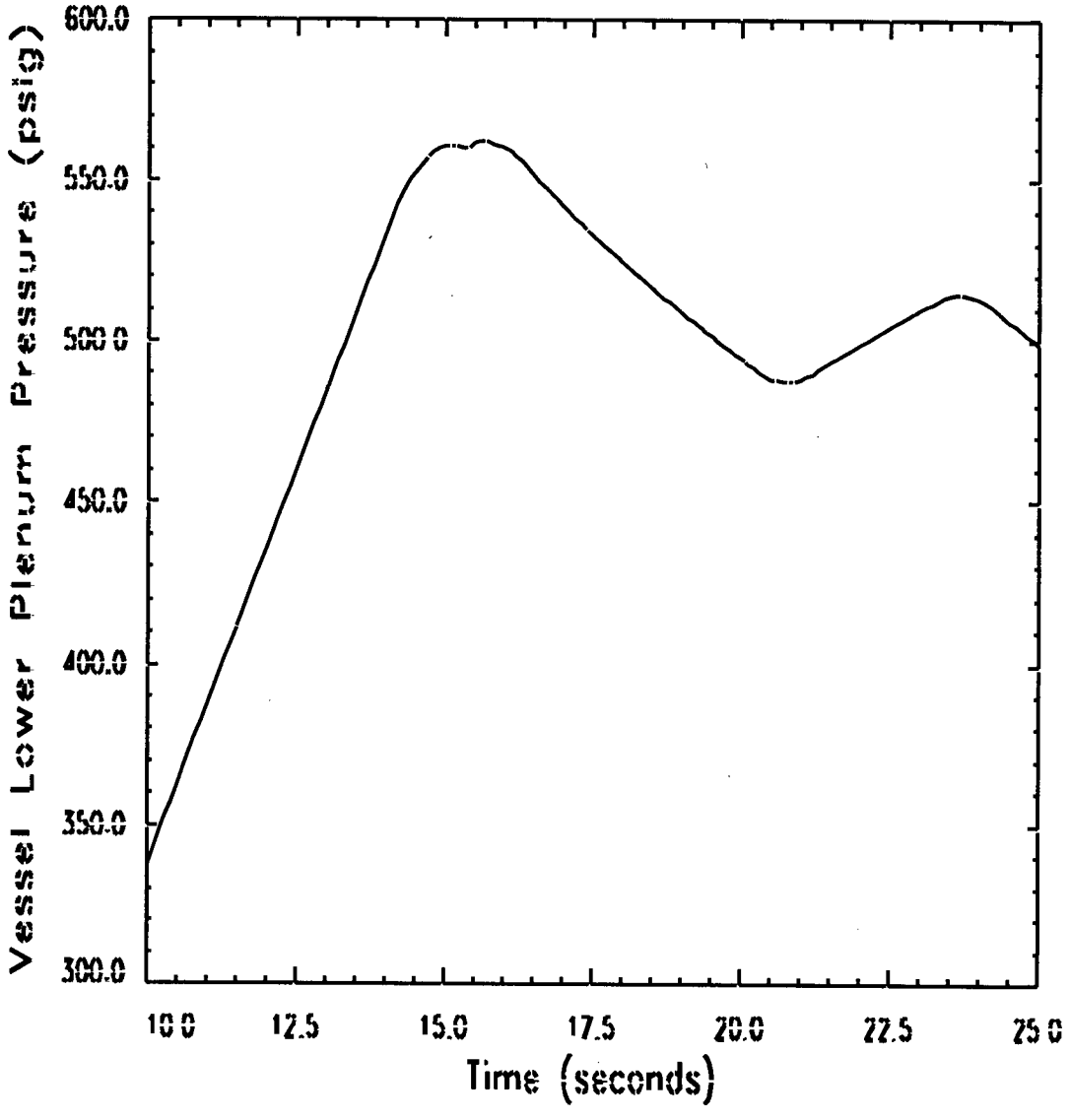

Figure 3.1 Pressure in Vessel Lower Plenum (Limiting Case)

4.0 REFERENCES

1. "Technical Specifications and Bases for H.B. Robinson Unit No. 2," Appendix A to the Facility Operating License DPR-23, CP&L Docket No. 50-261.

2. "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," ANF-89-151 (P) (A), Siemens Nuclear Power Corporation, April 1992.

3. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, LWR Edition, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, July 1981.

4. CP&L Letter to NRC, "Reactor Coolant Overpressurization Pressure Transient Analysis Results, dated July 28, 1977.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
CONVERSION PACKAGE SECTION 1.0

ITS CONVERSION PACKAGE

CHAPTER 1.0 - USE AND APPLICATION

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 1.0 - USE AND APPLICATION

PART 1

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

1.0

Use AND APPLICATION

~~TECHNICAL SPECIFICATIONS AND BASES~~

1.1 A1

1.~~0~~ 1 **DEFINITIONS**

NOTE

NOTE

The ~~following frequently~~ used ~~terms are defined for the uniform interpreta-tion of the specifications.~~ defined of this section appear in capitalized type and are applicable throughout these Techincal Specifications and Bases

~~1.1~~ Rated Thermal Power (RTP)

~~A steady state nuclear steam supply output (reactor core thermal power) of 2300 MWt.~~ RTP shall be a total reactor core heat transfer Rate to the reactor coolant of 2300 MWt A2

~~1.2 Reactor Operation~~

~~1.2.1 (Deleted by Change No. 21 issued 7/2/73)~~ A1

TABLE 1.1-1 ~~1.2.2~~ Cold Shutdown ~~Condition~~

When ~~the reactor is subcritical~~ and T_{avg} is ≤ 200°F. $K_{eff} < 0.99$ A4 A3

TABLE 1.1-1 ~~1.2.3~~ Hot Shutdown ~~Condition~~

When ~~the reactor is subcritical~~ and ~~T_{avg} is > 200°F.~~ 350°F > T_{avg} > 200°F

TABLE 1.1-1 ~~1.2.4~~ ~~Reactor Critical~~ Startup

When ~~the neutron chain reaction is self-sustaining~~ and k_{eff} ~~= 1.0~~ ≥ 0.99 and RTP ≤ 5%

A4

~~1.2.5~~ Power Operating ON ~~Condition~~

TABLE 1.1-1 When ~~the reactor is critical~~ and ~~the neutron instrumentation indicates greater than 2% rated power.~~ $K_{eff} \geq 0.99$ RTP > 5% A5

A3

Add table 1.1-1, MODE 3 - Hot Standby

Add Definitions:

ACTUATION LOGIC TEST
AXIAL FLUX DIFFERENCE (AFD)
LEAKAGE
MASTER RELAY TEST
MODE
PHYSICS TEST
SHUTDOWN MARGIN (SDM)
SLAVE RELAY TEST
THERMAL POWER
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)

A6

Add table 1.1-1 Note(a) A5

Add table 1.1-1 Notes (b) and (c) A7

Chapter 1.1

CORE ALTERATION

A1

1.2.6 Refueling Operation

Fuel sources or reactivity control components within the reactor vessel

Any operation involving movement of core components when there is fuel in the containment vessel and the pressure vessel head is unbolted or removed.

A7

Suspension of core Alterations shall not preclude completion of movement of a component to a safe position

1.2.7 Operating Basis Earthquake

The operating basis earthquake is that earthquake which involves a ground acceleration of 0.10 g horizontally and 0.067 g vertically.

A8

1.2.8 Safe Shutdown Earthquake

The safe shutdown earthquake is that earthquake which involves a ground acceleration of 0.20 g horizontally and 0.133 g vertically.

A9

1.3 OPERABLE - OPERABILITY

Safety

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). and when

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

or

Specified Safety

and

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

See 3.8

1.4 PROTECTION INSTRUMENTATION CHANNEL

An arrangement of components and modules are required to generate a single protective action signal when required by a plant condition. A channel loses its identity where single action signals are combined.

A8

1.5 DEGREE OF REDUNDANCY

The difference between the number of operable channels and the number of channels which when tripped will cause an automatic system trip.

Add TABLE 1.1-1 MODE 6 - Refueling

A7

A1

~~1.0 INSTRUMENTATION SURVEILLANCE~~

~~1.0.1~~ Action~~s~~

Action shall be that part of a specification ~~which~~ that prescribes ~~remedial measures~~ required ~~under designated conditions~~ actions to be taken under designated conditions within specified completion times.

~~1.0.2~~ Channel Calibration

Adjustment of the channel ~~output~~ such that it responds, with ~~acceptable~~ required range and accuracy, to known ~~value of the parameter which the channel measures~~ input. Calibration shall encompass the entire channel, including ~~the~~ alarm ~~or trip~~, required sensor and shall be deemed to include the channel functional test. interlock, display and trip functions

A10

~~1.0.3~~ Channel Check

Insert C1.1-1

A qualitative ~~determination of acceptable operability~~ assessment, by observation of channel behavior during operation. This determination will include, whenever possible, comparison of the channel indication and status with other independent channels measuring the same variable.

A13

~~1.0.4~~ Channel ~~Functional~~ operational Test (COT)

Injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify that it is operable, including required alarm, interlock, display and/~~or~~ trip initiating action.

Insert C1.1-2

1.0.5 Source Check

A8

A source check shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

1.7 CONTAINMENT INTEGRITY

Containment integrity is defined to exist when:

See 3.6.1
3.6.2
3.6.3

Insert C1.1-1

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

INSERT C1.1-2

The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.

A1

a. All non-automatic containment isolation valves not required for normal operation are closed and blind flanges are properly installed where required.

b. The equipment door is properly closed and sealed.

c. At least one door in the personnel air lock is properly closed and sealed.

d. All automatic containment isolation trip valves required to be closed during accident conditions are operable or are secured closed except as stated in Specification 3.6.3. Manual valves qualifying as automatic containment isolation valves are secured closed.

e. The uncontrolled containment leakage satisfies Specification 4.4.

See 3.6.1 3.6.2 3.6.3

~~1.8~~ QUADRANT POWER TILT RATIO (QTPR)

upper excore detector calibrated output

The quadrant power tilt is defined as the ratio of maximum to average of the upper excore detector ~~currents~~ or the ~~lower excore detector currents~~, whichever is greater. If one excore is out of service, the three in-service units are used in computing the average.

calibrated outputs

see 3.2.4

SR 3.2.4.1 NOTE 1

1.9 DELETED

the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs

~~1.10~~ STAGGERED TEST BASIS

A Staggered Test Basis shall consist of:

~~a. A test schedule for n systems, subsystems, trains or designated components obtained by dividing the specified test interval into n equal subintervals.~~

A11

testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n surveillance frequency intervals, where n is the total number of systems, subsystems, channels or other designated components in the associated function.

Chapter 1.1

A1

~~b. The testing of one system, subsystem, train or designated components at the beginning of each subinterval.~~

1.11 GASEOUS RADWASTE TREATMENT SYSTEM

The Gaseous Radwaste Treatment System is the system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

1.12 VENTILATION EXHAUST TREATMENT SYSTEM

The Ventilation Exhaust Treatment System is the system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters prior to their release to the environment. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be Ventilation Exhaust Treatment System components.

A8

1.13 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The Offsite Dose Calculation Manual shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and the methodology to calculate gaseous and liquid effluent monitoring alarm/trip setpoints; and, the requirements of the environmental radiological monitoring program.

see 5.5.1

~~1.14~~ DOSE EQUIVALENT I-131

The Dose Equivalent I-131 shall be that concentration of I-131 (microcurie/gram) ~~which~~ that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and

~~Amendment No. 85~~

I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in NRC Regulatory Guide 1.109, Revision 1, October 1977.

1.15 PROCESS CONTROL PROGRAM (PCP)

The Process Control Program (PCP) shall contain the current formulas, sampling, analyses, tests and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71, and Federal and State regulations and other requirements governing the disposal of the radioactive waste.

See 5.5.1

1.16 SOLIDIFICATION

Solidification shall be the conversion of wet radioactive wastes into a form that meets shipping and burial ground requirements.

(A8)

1.17 PURGE - PURGING

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

1.18 VENTING

Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during venting. Vent, used in system names, does not imply a venting process.

Chapter 1.1

A1

1.19 SITE BOUNDARY

The site boundary shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee, as defined by Figure 1.1-1.

see 4.1

1.20 MEMBER(S) OF THE PUBLIC

Member(s) of the public shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen, or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for the purposes of protection of individuals from exposure to radiation and radioactive materials.

A8

1.21 UNRESTRICTED AREA

Unrestricted area shall be any area at or beyond the Site Boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the Site Boundary used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

~~1.22~~ CORE OPERATING LIMITS REPORT (COLR)

The ~~CORE OPERATING LIMITS REPORT (COLR)~~ is the unit-specific document that provides ~~core operating~~ limits for the current ~~operating~~ reload cycle. These cycle-specific ~~core operating~~ limits shall be determined for each reload cycle in accordance with Specification ~~6.9.3.3.~~ ~~Unit~~ operation within these operating limits is addressed in individual specifications.

cycle specific parameter

5.6.5

Plant

parameter

Add

1.2 LOGICAL CONNECTORS
1.3 COMPLETION TIMES
1.4 FREQUENCY

A12

Chapter 1.1

A1

3.1.4 Maximum Reactor Coolant Activity

See 3.4.16

The total specific activity in Ci/gram of the reactor coolant shall not exceed 1.0 Ci/gram dose equivalent I-131 and 100/$\bar{E}$ Ci/gram under all modes of operation. *$\bar{E}$ is the average beta and gamma energy (MEV) per disintegration of the specific activity.*

Insert C1.1-3

Insert C.1.1-4

M1

With the specific activity of the primary coolant > 1.0 Ci/gram dose equivalent I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.1.4-1, be in at least hot shutdown with T_{avg} < 500°F within 6 hours.

With the specific activity of the primary coolant > 100/$\bar{E}$ Ci/gram, be in at least hot shutdown with T_{avg} < 500°F within 6 hours.

See 3.4.16

In any operating mode, with the specific activity of the primary coolant > 1.0 Ci/gram dose equivalent I-131 or > 100/$\bar{E}$ Ci/gram, perform the sampling and analysis requirements of Item 1 of Table 4.1-2 until the specific activity of the primary coolant is restored to within its limits.

Insert C1.1-3

(weighted in proportion to the concentrations of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average

Insert C1.1-4

for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 1.0 - USE AND APPLICATION

PART 2

DISCUSSION OF CHANGES (DOCS) FOR CTS MARKUP

ADMINISTRATIVE CHANGES

A1 In the conversion of the H.B. Robinson Steam Electric Plant (HBRSEP), Unit 2 Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in the - Standard Technical Specifications, Westinghouse Plants NUREG-1431, Rev 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 The CTS definition of Rated Power can be interpreted to refer to both a steady state nuclear steam supply output and reactor core thermal power. The ITS definition includes only reactor core thermal power, consistent with ISTS. This is an administrative change which provides clarification consistent with the CTS's parenthetical reference to reactor core thermal power and the Operating License, Condition 3.A, specification of Maximum Power Level as reactor power level. Therefore, this change has no adverse impact on safety.

A3 The CTS definition of Hot Shutdown Condition includes the condition with the reactor subcritical and reactor coolant average temperature (T_{avg}) greater than 200°F. The ITS has two MODEs, ITS MODE 3 - Hot Standby and ITS MODE 4 - Hot Shutdown, which collectively cover the same condition. The separation of the CTS Hot Shutdown Condition into the two ITS MODEs is an administrative change to establish consistency with the ISTS, which has no adverse impact on safety. The specific impact of the definition of MODES 3 and 4 in lieu of the Hot Shutdown Condition is discussed with the applicable LCOs when the change results in a more or less restrictive change.

A4 The CTS definitions for Hot Shutdown Condition, Cold Shutdown Condition, Reactor Critical and Power Operation are specified in terms of the reactor being subcritical (K_{eff} <1.0) or critical ($K_{eff} \geq 1.0$). The related ITS MODEs 1 through 5 are specified in terms of K_{eff} either being < 0.99 or ≥ 0.99. Since changing from CTS definitions of operating conditions to ITS definitions of MODEs is not, in itself, either more or less restrictive, this is an administrative change which establishes consistency with ISTS MODE definitions. The specific impact of the utilization of the definitions of MODEs 1 through 5 on the related CTS Condition is discussed with the applicable LCOs when the utilization results in a more or less restrictive change.

A5 The CTS definition for Power Operating Condition is specified as, "When the reactor is critical and the neutron instrumentation indicates greater than 2% rated power." The CTS definition for Reactor Critical is specified as, "When the neutron chain reaction is self sustaining and

K_{eff} = 1.0" without any specification regarding power level. The ITS translates these two operational conditions to MODEs 1 and 2 respectively. MODE 1 is specified in ITS as $K_{eff} \geq 0.99$ and % RATED THERMAL POWER > 5% while MODE 2 is specified in ITS as $K_{eff} \geq 0.99$ and % RATED THERMAL POWER ≤ 5%. The ITS adds a footnote to state that the RATED THERMAL POWER limits in Table 1.1-1 exclude decay heat.

The ITS is silent regarding the specific method to measure reactor power. Since neutron instrumentation is normalized to thermal power calculations, the lack of specificity regarding the measurement of reactor power is inconsequential and thus has no adverse impact on safety. The change from 2% to 5% RTP less decay heat will permit operation at a greater power level prior to entry into ITS MODE 1 than permitted by CTS prior to entry into the Power Operating Condition. The specific impact of the change regarding the ITS MODE 1 and 2 definitions is evaluated for each relevant LCO. Therefore, this change has no adverse impact on safety.

A6 During the ITS development certain definitions which are not part of the CTS are adopted from the ISTS. The definitions are:

ACTUATION LOGIC TEST	AXIAL FLUX DIFFERENCE (AFD)
LEAKAGE	MASTER RELAY TEST
MODE	PHYSICS TESTS
SHUTDOWN MARGIN (SDM)	SLAVE RELAY TEST
THERMAL POWER	TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)

The adoption of these definitions results in no technical changes (either actual or interpretational) to the CTS. Therefore, this is an administrative change and has no adverse impact on safety.

A7 The CTS defines Refueling Operation as, "Any operation involving movement of core components when there is fuel in the containment vessel and the pressure vessel head is unbolted or removed." The ITS definition for CORE ALTERATION is the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The ITS definition for CORE ALTERATION adds additional details regarding what constitutes "movement of core components" and excludes the condition when the head is unbolted (but not removed). Additionally, the ITS adds the definition for MODE 6 - Refueling, including footnotes (b) and (c) to Table 1.1-1 regarding tensioning of the reactor vessel head closure bolts. No definition comparable to ITS MODE 6 exists in the CTS.

Since these changes are not, in themselves, either more or less restrictive, they are administrative changes to establish consistency with the ISTS definition. The specific impact of the utilization of the

definition of CORE ALTERATION is discussed with the applicable LCOs when the change results in a more or less restrictive change. The specific impact of the utilization of the definition of MODE 6 is discussed with the applicable LCOs when the change results in a more or less restrictive change.

A8 Selected CTS definitions are deleted because the CTS that use these definitions are not retained in the ITS or the equivalent ITS Specification will not use the defined term. Discussions of the technical aspects of these changes are addressed in the discussion of changes for the specifications where the phrase is used in the ITS. The removal of a definition that is not used in the ITS is an administrative change because it has no impact on the implementation of any existing requirement not addressed in the ITS development and has no adverse impact on safety.

A9 The first paragraph of the CTS definition of OPERABLE - OPERABILITY requires the availability of, "all necessary attendant instrumentation, controls, normal <u>and</u> emergency electrical power." The ITS definition specify, "all necessary attendant instrumentation, controls, normal <u>or</u> emergency electrical power." This is an administrative change because the second paragraph to the CTS definition requires only one source to be operable as long as the redundant systems, subsystems, trains components, and devices are OPERABLE. The second paragraph to the CTS definition is incorporated into ITS Section 3.8.1. (Appropriate discussion of the utilization of the second paragraph in ITS Section 3.8 is provided with ITS Section 3.8.) Thus, the ITS requirements are essentially the same as the CTS. Therefore, there is no adverse impact on safety.

A10 The CTS definition of Channel Calibration does not exclude RTD or thermocouples from the generic description of calibration encompassing the entire channel. The ITS recognize the nature of these devices as non-adjustable and permits the calibration of RTD and thermocouple channels using a qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Although RTD and thermocouple sensors can have their input/output response characteristics confirmed, these devices are not adjustable and not likely to drift outside of an acceptable range. The specific impact of this change is evaluated with the applicable surveillance requirements. Therefore, this is an administrative change and has no adverse impact on safety.

A11 The CTS definition of Staggered Test Basis requires testing <u>N</u> systems, subsystems, channels or other designated components within one surveillance interval where <u>N</u> is the total number of systems, subsystems, channels or other designated components. The ITS specifies testing <u>N</u> systems, systems, channels or other designated components within <u>N</u> surveillance intervals. The impact of the change in the

definition is evaluated for each applicable surveillance. Therefore, this is an administrative change with no adverse impact on safety. This is consistent with the ISTS definition.

A12 The CTS does not include discussions of Logical Connectors, Completion Times and Frequency. These discussions have been included in the ITS to aid in the understanding and use of the ITS. Some conventions in applying the Technical Specifications to unique situations have previously been the subject of differing interpretations. The guidance in the ITS is consistent with the ISTS and does not conflict or deviate from the CTS. Therefore these changes are considered administrative and have no adverse impact on safety.

A13 The CTS definition of Channel Functional Test states, "Injection of a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating actions." The CTS definition of CHANNEL OPERATIONAL TEST (COT) states, "A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy."

Use of an actual signal in lieu of a simulated signal can provide some flexibility in actual performance of testing. However, the use of an actual signal in lieu of a simulated signal constitutes an equally valid methodology for testing a channel and does not result in a change to the results of the test. Consequentially, this aspect of the change is administrative.

The COT requires, ". . . injection of the signal into the channel as close to the sensor as practical . . ." Although the CTS definition for Channel Functional Test is not explicit regarding the point of test signal injection, inclusion of the sensor in the test is implied when such inclusion is practical. When including the sensor is practical, it is generally the point of signal injection. Consequentially, this aspect of this change is administrative.

The COT explicitly includes required interlock and display functions. Although not explicitly stated, the CTS definition for Channel Functional Test encompasses these <u>required</u> interlock and display functions by requiring verification that the channel is OPERABLE.

The COT explicitly requires adjustment, as necessary, of <u>required</u> alarms, interlocks and trip setpoints so the setpoints are within the <u>required</u> range and accuracy. Although not explicitly stated, the CTS definition for Channel Functional Test encompasses this requirement by requiring verification that the channel is OPERABLE. Implicit in the

CTS is the requirement to enter appropriate CTS action statements for CTS surveillance tests with out-of-tolerance results. Entry into an action statement requires repair (e.g., adjustment) within a limited time or otherwise comply with the CTS action statement. These are administrative changes which provide additional detail and are consistent with ISTS.

A14 CTS 3.1.4 provides a definition of $\bar{E}$. The ITS definition provides additional details regarding the determination of E. The ITS definition is consistent the CTS definition as clarified by a CTS bases description which existed prior to CTS Amendment No. 108. Although the additional descriptive information in the CTS bases was eliminated by this amendment, neither the application for the associated Technical Specification revision (CP&L letter to NRC dated 1/8/96, Request for License Amendment, Reporting Requirements for Primary Coolant Iodine Spiking) nor the Safety Evaluation Report (SER) for the amendment (issued 10/28/86) identified any technical significance for the elimination of the descriptive detail from the CTS Bases. This is an administrative change which provide additional detail and is consistent with ISTS.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS Specification 3.1.4, which defines $\bar{E}$ as the average of beta and gamma energy disintegration of the specific activity is revised in the ITS definition to include weighted average and the composition of isotopes to exclude iodine. Since this change adds requirements to the CTS that are currently found only in procedures, this change is more restrictive, and has no adverse impact on safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

None

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

None

SPECIFICATIONS RELOCATED

None

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 1.0 - USE AND APPLICATION

PART 3

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC), AND BASIS FOR CATEGORICAL EXCLUSION FROM 10 CFR 51.22

ADMINISTRATIVE CHANGES
("A" Labeled Comments/Discussions)

In the conversion of the HBRSEP Unit No. 2 Technical Specifications to the proposed plant specific Improved Technical Specifications certain wording preferences or conventions are being adopted which do not result in technical changes (either actual or interpretational). Editorial changes, clarification, reformatting, rewording and revised numbering are being adopted to make the improved Technical Specifications consistent with NUREG 1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Administrative" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes consist of editorial changes and clarification, reformatting, rewording and renumbering of the current Technical Specifications. This process does not involve any technical changes to existing requirements. As such, these changes are administrative in nature and do not impact initiators of analyzed events or alter any assumptions relative to mitigation of accident or transient events. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. The proposed changes do not impose or eliminate any requirements. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed changes will not reduce a margin of safety because they do not impact any safety analysis assumptions. These changes are administrative in nature and, as such, do not impact any technical requirements. Therefore, these changes do not involve any reduction in a margin of safety.

MORE RESTRICTIVE CHANGES
("M" Labeled Comments/Discussions)

The HBRSEP Unit No. 2 Technical Specifications are proposed to be modified in some areas to impose more restrictive requirements than currently exist. These more restrictive changes are being imposed to be consistent with NUREG 1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "More Restrictive" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes provide requirements determined to be more restrictive than the current Technical Specifications requirements for operation of the facility. These more restrictive requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. These changes have been confirmed to ensure that no previously evaluated accident has been adversely affected. The more restrictive requirements being proposed enhance assurance that process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis of the unit. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. These changes do impose new or additional requirements which are consistent with assumptions made in the safety analysis and licensing basis. The additional requirements include new Surveillance Requirements, more restrictive Frequencies and Completion Times, new LCOs, more restrictive Required Actions and Applicabilities, and other operational restrictions that enhance safe operation. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact or increases the margin of plant safety. Each of the changes in this category, while providing new or additional requirements designed to

enhance plant safety, is consistent with the safety analyses and licensing basis. Therefore, these changes do not involve a reduction in a margin of safety.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
NRC DOCKET NO. 50-261/LICENSE NO. DPR-23
REQUEST FOR TECHNICAL SPECIFICATION CHANGE
CONVERSION TO IMPROVED STANDARD TECHNICAL SPECIFICATIONS

ENVIRONMENTAL CONSIDERATIONS

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulator actions for categorical exclusion for performing an environmental assessment. A proposed change for an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; (3) result in an increase in individual or cumulative occupational radiation exposure. We have reviewed this request and determined that the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance with the amendment. The basis for this determination follows.

Proposed Change

This request proposes to change the technical specifications to be consistent with NUREG-1431; Standard Technical Specifications, Westinghouse Plants Revision 1, 04/07/95 within limitations imposed by plant specific design and licensing basis.

Basis

The proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons.

1. As demonstrated in the No Significant Hazards Evaluation, the proposed changes do not involve a significant hazards consideration.

2. These proposed changes are being made to establish consistency with the improved Standard Technical Specifications (ISTS) - Westinghouse Plants, NUREG 1431, Rev. 1, including approved generic changes, and do not involve physical changes to the facility, nor do they affect actual plant effluents.

3. These proposed changes are being made to establish consistency with the improved Standard Technical Specifications (ISTS) - Westinghouse Plants, NUREG 1431, Rev. 1, including approved generic changes and do not involve physical changes to the facility, and they do not significantly affect individual or cumulative occupational radiation exposures.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 1.0 - USE AND APPLICATION

PART 4

MARKUP OF NUREG 1431, REVISION 1, "STANDARD TECHNICAL SPECIFICATIONS - WESTINGHOUSE PLANTS" (ISTS)

CTS

1.1 Definitions (continued)

[1.6.3] **CHANNEL CHECK**

A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

[1.6.4] **CHANNEL OPERATIONAL TEST (COT)**

A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.

[1.2.6] **CORE ALTERATION**

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

[1.22] **CORE OPERATING LIMITS REPORT (COLR)**

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

[1.14] **DOSE EQUIVALENT I-131**

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page

(continued)

CTS

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	192.212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity"].
[3.1.4] Ē-AVERAGE DISINTEGRATION ENERGY	Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > ~~15~~ minutes, making up at least 95% of the total noniodine activity in the coolant.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. (2)
L_a	The maximum allowable primary containment leakage rate, L_a, shall be [ ]% of primary containment air weight per day at the calculated peak containment pressure (P_a). (4)
[Doc A6] LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or ~~leakoff~~ return), that is captured and conducted to collection systems or a sump or collecting tank; (1)

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection

(continued)

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1.1 Definitions

[Doc A6] LEAKAGE (continued)

systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or ~~leakoff~~ return) that is not identified LEAKAGE; 1

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

[Doc A6] MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

[Doc A6] MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

[1.3] OPERABLE – OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

[Doc A6] PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

(continued)

CTS

1.1 Definitions

[Doc A6]

PHYSICS TESTS (continued)

a. Described in Chapter ~~X~~14, Initial Test Program~~X~~ of the ~~FSAR~~; Updated Final Safety Analysis Report (UFSAR) (1)

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

~~**PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)**~~

~~The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."~~ (3)

[1.8]

QUADRANT POWER TILT RATIO (QPTR)

QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

[1.1]

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~[2893]~~ 2300 MWt.

~~**REACTOR TRIP SYSTEM (RTS) RESPONSE TIME**~~

~~The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~ (2)

[Doc A6]

SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

(continued)

CTS

1.1 Definitions

[Doc A6] SHUTDOWN MARGIN (SDM) (continued)

a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the [nominal zero power design level] 547 °F

[Doc A6] SLAVE RELAY TEST

A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.

[1.10] STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

[Doc A6] THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

[Doc A6] TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)

A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, display, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

Table 1.1-1 (page 1 of 1)
MODES

CTS

	MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
[1.2.5]	1	Power Operation	≥ 0.99	> 5	NA
[1.2.4, 1.2.5]	2	Startup	≥ 0.99	≤ 5	NA
[1.2.3]	3	Hot Standby	< 0.99	NA	≥ [350]
[1.2.3]	4	Hot Shutdown[b]	< 0.99	NA	[350] > T_{avg} > [200]
[1.2.2]	5	Cold Shutdown[b]	< 0.99	NA	≤ [200]
[DOC A7]	6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

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

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

CTS

[Doc A12]

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

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

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

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

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

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

CTS
[DOC A12]

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability. If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition. Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

(continued)

1.3 Completion Times

DESCRIPTION (continued)

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

a. Must exist concurrent with the first inoperability; and

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

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

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

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

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

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

(continued)

1.3 Completion Times (continued)

EXAMPLES

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

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

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

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

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

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

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

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

Condition A has expired. LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

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

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

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

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

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

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

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

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

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

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

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

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

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

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

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-5 (continued)

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

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

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x.	Once per 8 hours
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-6 (continued)

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

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

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

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

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time

(continued)

1.3 Completion Times

EXAMPLES	EXAMPLE 1.3-7 (continued) Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.
IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

CTS
[Doc A12]

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

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

DESCRIPTION

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

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

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

EXAMPLES

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

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

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

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

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

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

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

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

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

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

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

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

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

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 1.0 - USE AND APPLICATION

PART 5

JUSTIFICATION FOR DIFFERENCES (JFDs) TO ISTS

1. In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes which involve the insertion of plant specific terms or parameters are used to preserve consistency with the CTS and licensing basis.

2. The definitions of ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME (Page 1.1-3) and REACTOR TRIP SYSTEM (RTS) RESPONSE TIME (Page 1.1-5) are not incorporated into the ITS because such response time testing is not part of the current plant licensing basis.

3. The definition for Pressure Temperature Limits Report (PTLR) is not utilized in the ITS. Pressure/Temperature limits in CTS Figure 3.1-1, Reactor Coolant System Heatup Limitations and CTS Figure 3.1-2, Reactor Coolant System Cooldown Limitations are retained in the ITS. ISTS references to the PTLR are modified in the ITS to refer to these figures.

4. The definition for L_a is not adopted in the ITS. The implementation of 10 CFR 50 Appendix J, Option B has resulted in modifications to the ISTS which capture the Leakage Rate Testing Program requirements in ITS paragraph 5.5.16. ITS section 5.5.16 provides a description of L_a which is consistent with the definition of L_a in ISTS Section 1.1, Definitions.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 1.0 - USE AND APPLICATION

PART 6

MARKUP OF ISTS BASES

N/A

CHAPTER 1.0 - USE AND APPLICATION DOES NOT INCLUDE A BASES SECTION

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 1.0 - USE AND APPLICATION

PART 7

JUSTIFICATION FOR DIFFERENCES (JFDs) TO ISTS BASES

N/A

CHAPTER 1.0 - USE AND APPLICATION DOES NOT INCLUDE A BASES SECTION

APPENDIX A

TO

THE FACILITY OPERATING LICENSE DPR-23

TECHNICAL SPECIFICATIONS

FOR

H. B. ROBINSON STEAM ELECTRIC PLANT

UNIT NO. 2

CAROLINA POWER & LIGHT COMPANY

DARLINGTON COUNTY, S.C.

DOCKET NO. 50-261

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BASES

TO

THE FACILITY OPERATING LICENSE DPR-23

TECHNICAL SPECIFICATIONS

FOR

H. B. ROBINSON STEAM ELECTRIC PLANT

UNIT NO. 2

CAROLINA POWER & LIGHT COMPANY

DARLINGTON COUNTY, S.C.

DOCKET NO. 50-261

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1.0 USE AND APPLICATION

1.1 Definitions

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

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, interlock, display, and trip functions. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.

(continued)

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Regulatory Guide 1.109, Rev. 1, NRC, 1977.
$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than

(continued)

1.1 Definitions

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY (continued)	iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or return), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System; b. <u>Unidentified LEAKAGE</u> All LEAKAGE (except RCP seal water injection or return) that is not identified LEAKAGE; c. <u>Pressure Boundary LEAKAGE</u> LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning

(continued)

1.1 Definitions

MODE (continued)	specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the Updated Final Safety Analysis Report (UFSAR); b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2300 MWt.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

(continued)

1.1 Definitions

SHUTDOWN MARGIN (continued)	a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the 547°F.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, display, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[b]	< 0.99	NA	350 > T_{avg} > 200
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

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

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

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

(continued)

1.2 Logical Connectors (continued)

EXAMPLES

The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

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

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

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

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

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

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability. If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition. Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

(continued)

1.3 Completion Times

DESCRIPTION (continued)

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

a. Must exist concurrent with the first inoperability; and

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

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

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

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

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

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

(continued)

1.3 Completion Times (continued)

EXAMPLES

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

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

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

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

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

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

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

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

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

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

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

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

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

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

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

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

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

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

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

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

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

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

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-5 (continued)

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

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

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x.	Once per 8 hours
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-6 (continued)

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

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

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

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

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-7 (continued)

Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

IMMEDIATE COMPLETION TIME

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

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.
EXAMPLES	The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

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

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

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

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

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

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

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

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

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

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

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

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 1.0 - USE AND APPLICATION

PART 9

PROPOSED BASES TO HBRSEP, UNIT NO. 2 ITS

N/A

CHAPTER 1.0 - USE AND APPLICATION DOES NOT INCLUDE A BASES SECTION

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 1.0 - USE AND APPLICATION

PART 10

ISTS GENERIC CHANGES

N/A

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
CONVERSION PACKAGE SECTION 2.0

ITS CONVERSION PACKAGE

CHAPTER 2.0 - SAFETY LIMITS (SLs)

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 2.0 - SAFETY LIMITS (SLs)

PART 1

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

Specification 2.1

(SLs)

A1

ITS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

SLs

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, coolant temperature, and flow when the reactor is critical.

M1

IN MODES 1+2

Objective

To maintain the integrity of the fuel cladding.

A2

System highest cold leg

Specification

[2.1.1]

a. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1 when full flow from three reactor coolant pumps exists.

and pressurizer

b. When full flow from one reactor coolant pump exists, the thermal power level shall not exceed 20%, the coolant pressure shall remain between 1820 psig and 2400 psig, and the Reactor Coolant System average temperature shall not exceed 590°F.

c. When natural circulation exists, the thermal power level shall not exceed 12%, the coolant pressure shall remain between 2135 psig and 2400 psig, and the Reactor Coolant System average temperature shall not exceed 620°F.

See 3.4.4

d. The safety limit is exceeded if the combination of Reactor Vessel inlet temperature and thermal power level is at any time above the appropriate pressure line in Figure 2.1-1 or if the thermal power level, coolant pressure, or Reactor Vessel inlet temperature violates the limits specified above.

A3

Specification 2.1

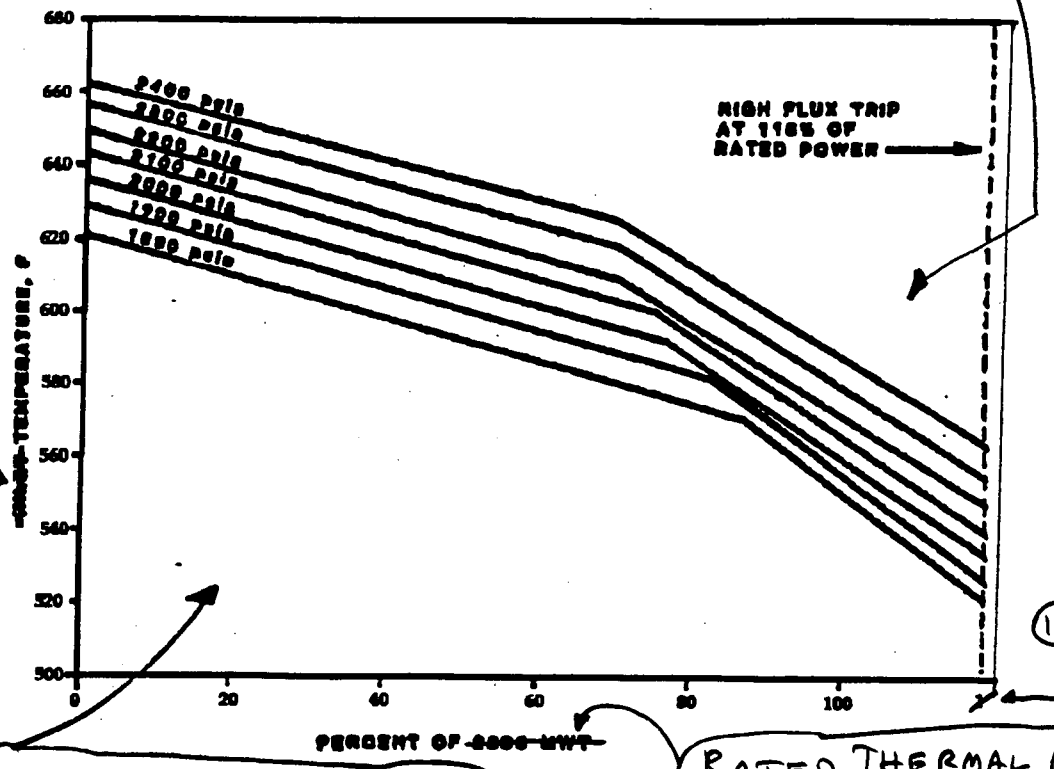
A1

A2

RCS HIGHEST COLD LEG

DO NOT OPERATE IN THIS AREA

A4



120

RATED THERMAL POWER

ACCEPTABLE OPERATION

A4

CORE PROTECTION BOUNDARIES FOR 3-LOOP OPERATION
Figure 2.1-1

NOTE: BASED ON A MINIMUM RCS FLOW OF 97.3×10^6 lbm/hr

Specification 2.1

A1

2.2 SAFETY LIMIT, REACTOR COOLANT SYSTEM PRESSURE

Applicability

Applies to the limit on Reactor Coolant System pressure.

Objective

To maintain the integrity of the Reactor Coolant System and to prevent the release of excessive amounts of fission product activity to the coolant.

Specification

In MODES 1, 2, 3, 4 and 5 L1

ITS [2.1.2]

The Reactor Coolant System pressure shall be maintained L ~~not exceed~~ 2735 psig ~~with fuel assemblies installed in the reactor vessel~~.

Basis

The Reactor Coolant System[1] serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the Reactor Coolant System is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the reactor coolant system and fuel cladding is assured. The maximum transient pressure allowable in the Reactor Coolant System pressure vessel under the ASME Code, Section III is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USAS Section B31.1 is 120% of design pressure. Thus, the safety limit of 2735 psig (110% of design pressure) has been established.[2]

The settings of the power-operated relief valves (2335 psig), the reactor high pressure trip (2385 psig) and the safety valves (2485 psig) have been established to assure never reaching the Reactor Coolant System pressure safety limit. The initial hydrostatic test is conducted at 3110 psig to assure the integrity of the Reactor Coolant System.

A7

Reference

(1) FSAR Section 4
(2) FSAR Section 4.3

Specification 2.2

ITS 2.2 SL

A1

6.7 SAFETY LIMIT VIOLATION

[2.2] 6.7.1 The following actions shall be taken in the event a safety limit is violated:

a) The provisions of 10 CFR 50.72 shall be complied with. A5

b) The provisions of 10 CFR 50.36(c)(1)(i) shall be complied with.

c) The safety limit violation shall be reported to the NRC Region II within one hour and the Vice President - Robinson Nuclear Plant within 24 hours.

d) A Safety Limit Report shall be prepared. The report shall be reviewed in accordance with Specification 6.5.1.6.6. This report shall describe (1) applicable circumstances preceding the violation; (2) effects of the violation upon facility components, systems, or structures; and (3) corrective action taken to prevent recurrence. LA1 A6

e) The Safety Limit Violation Report shall be submitted to the NRC, Vice President - Robinson Nuclear Plant, and the Manager - Nuclear Assessment Section within 14 days of the violation.

Add 2.2.1
Add 2.2.2 M2

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 2.0 - SAFETY LIMITS (SLs)

PART 2

DISCUSSION OF CHANGES (DOCS) FOR CTS MARKUP

ADMINISTRATIVE CHANGES

A1 In the conversion of the H.B. Robinson Steam Electric Plant (HBRSEP), Unit No.2 Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in the - Standard Technical Specifications, Westinghouse Plants NUREG-1431, Rev 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 The CTS 2.1 reactor core SL Specification is not stated in terms of a specific point for the temperature measurement, however the referenced CTS Figure 2.1-1 is specified in terms of inlet (i.e., cold leg) temperature. The ITS specifically describes the SL in terms of the highest RCS cold leg temperature. The addition of the specific reference to ". . . highest cold leg temperature . . ." is an administrative detail added for clarity.

A3 CTS 2.1.d is not retained in the ITS. CTS 2.1.d provides administrative details defining when the SLs specified in Figure 2.1-1 and in CTS 2.1.b and 2.1.c are exceeded. This type of clarification is included in the comparable ITS Figure 2.1.1-1. See A4 for additional information. The additional administrative detail regarding CTS 2.1.b and 2.1.c merely reiterate that violating the specified limits is exceeding the SLs. This change is consistent with NUREG-1431.

A4 CTS Figure 2.1-1 does not clearly define the regions of acceptable and unacceptable conditions. Administrative clarifications are included in ITS to delineate the appropriate regions on the figure. This change is consistent with NUREG-1431.

A5 CTS 6.7.1.a specifies that in the event of a SL violation, notification be made in compliance with 10 CFR 50.72. CTS 6.7.1.b specifies compliance with 10 CFR 50.36(c)(i). ITS does not retain these specifications. This change deletes requirements from the Technical Specifications that are duplicative of other regulations. Compliance with applicable regulations is required by the Operating license. Consequentially, this is an administrative change. This change is consistent with NUREG-1431, as modified by TSTF-5.

A6 CTS 6.7.1.c, in the event of a SL violation, specifies notification of NRC Region II within 1 hour. ITS does not retain this specification. 10 CFR 50.72 requires notification of the NRC Operations Center within 1 hour for events requiring declaration of an Emergency Classification, which include SL violations. This change deletes requirements from the Technical Specifications that are duplicative of other regulations.

Consequentially, this is an administrative change. This change is consistent with NUREG-1431 as modified by TSTF-5.

CTS 6.7.1.d, in the event of a SL violation, specifies preparation of a report and delineates the content and review requirement for this report. ITS does not retain this specification. 10 CFR 50.73(a)(2)(ii)(B) requires the submittal of a licensee event report (LER), for events which encompass safety limit violations and specifies content requirements. This change deletes requirements from the Technical Specifications that are duplicative of other regulations. Consequentially, this is an administrative change. This change is consistent with NUREG-1431 as modified by TSTF-5.

A7 The CTS Bases (and References) are not retained in the ITS, but are replaced in their entirety. The ITS includes significantly expanded and improved Bases. The Bases do not define or impose any specific requirements but serve to explain, clarify and document the reasons (i.e., Bases) for the associated Specification. The Bases are not part of the Technical Specifications required by 10 CFR 50.36. This change is administrative, and has no adverse impact on safety.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 The Applicability for the reactor core SL has been changed to not only include when the reactor is critical, but also when in MODE 2 and subcritical. This ensures that the SLs are met during reactor startup since their is a potential for a transient with the reactor near normal operating temperature and pressure conditions prior to criticality. This is an additional restriction on plant operation. This change is consistent with NUREG-1431.

M2 CTS 6.7.1.b specifies compliance with 10 CFR 50.36(c)(1)(i) in the event of a safety limit violation. This regulation requires the reactor be shutdown in the event of exceeding a SL, however this regulation does not explicitly require restoration of compliance with the SL and no time frames are delineated. ITS 2.2.1 and 2.2.2 specify, in the event of exceeding a SL while in MODE 1 or 2, ". . . restore compliance and be in MODE 3 within 1 hour." ITS 2.2.2 specifies, in the event of exceeding a SL while in MODE 3, 4 or 5, ". . . restore compliance within 5 minutes." The explicit requirement to restore compliance and the specified time limits are additional restrictions on plant operation. These changes are consistent with NUREG-1431.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS 6.7.1.c, in the event of a SL violation, specifies notification of the Vice President - Robinson Nuclear Plant within 24 hours. This detail is relocated to licensee procedures. This detail associated with the involved specification is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for compliance with the safety limits. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change to operational requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. This change is consistent with NUREG-1431 as modified by TSTF-5.

CTS 6.7.1.d, in the event of a SL violation, specifies review of the report by the Plant Nuclear Safety Committee (PNSC). This detail is relocated to licensee procedures. This detail associated with the involved specification is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for compliance with the safety limits. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. This change is consistent with NUREG-1431 as modified by TSTF-5.

CTS 6.7.1.e, in the event of a safety limit violation, specifies submittal of the report to the NRC, Vice President - Robinson Nuclear Plant and the Manager - Nuclear Assessment section within 14 days of the violation. This detail is relocated to licensee procedures. This detail associated with the involved specification is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for compliance with the safety limits. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. This change is consistent with NUREG-1431 as modified by TSTF-5.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 The CTS Applicability for the RCS pressure SL is specified "with fuel in the reactor vessel." The ITS Applicability is specified "MODES 1, 2, 3, 4, and 5." The ITS does not require the SL to be met with fuel in the vessel with one or more reactor vessel (RV) closure bolts less than fully tensioned or with the head removed. With one or more RV closure bolts less than fully tensioned, it is highly unlikely that the RCS can be pressurized greater than the SL pressure due to the low temperature over-pressure protection requirements. With the head removed, it is not possible to pressurize the RCS greater than the SL pressure. This change is consistent with NUREG-1431.

RELOCATED SPECIFICATIONS

None

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 2.0 - SAFETY LIMITS (SLs)

PART 3

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC), AND BASIS FOR CATEGORICAL EXCLUSION FROM 10 CFR 51.22

ADMINISTRATIVE CHANGES
("A" Labeled Comments/Discussions)

In the conversion of the HBRSEP Unit No. 2 Technical Specifications to the proposed plant specific Improved Technical Specifications certain wording preferences or conventions are being adopted which do not result in technical changes (either actual or interpretational). Editorial changes, clarification, reformatting, rewording and revised numbering are being adopted to make the improved Technical Specifications consistent with NUREG 1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Administrative" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes consist of editorial changes and clarification, reformatting, rewording and renumbering of the current Technical Specifications. This process does not involve any technical changes to existing requirements. As such, these changes are administrative in nature and do not impact initiators of analyzed events or alter any assumptions relative to mitigation of accident or transient events. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. The proposed changes do not impose or eliminate any requirements. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed changes will not reduce a margin of safety because they do not impact any safety analysis assumptions. These changes are administrative in nature and, as such, do not impact any technical requirements. Therefore, these changes do not involve any reduction in a margin of safety.

MORE RESTRICTIVE CHANGES
("M" Labeled Comments/Discussions)

The HBRSEP Unit No. 2 Technical Specifications are proposed to be modified in some areas to impose more restrictive requirements than currently exist. These more restrictive changes are being imposed to be consistent with NUREG 1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "More Restrictive" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes provide requirements determined to be more restrictive than the current Technical Specifications requirements for operation of the facility. These more restrictive requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. These changes have been confirmed to ensure that no previously evaluated accident has been adversely affected. The more restrictive requirements being proposed enhance assurance that process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis of the unit. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. These changes do impose new or additional requirements which are consistent with assumptions made in the safety analysis and licensing basis. The additional requirements include new Surveillance Requirements, more restrictive Frequencies and Completion Times, new LCOs, more restrictive Required Actions and Applicabilities, and other operational restrictions that enhance safe operation. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact or increases the margin of plant safety. Each of the changes in this category, while providing new or additional requirements designed to

enhance plant safety, is consistent with the safety analyses and licensing basis. Therefore, these changes do not involve a reduction in a margin of safety.

LESS RESTRICTIVE-GENERIC CHANGES
("LA" Labeled Comments/Discussions)

In the conversion of the HBRSEP Unit No. 2 Technical Specifications to the proposed plant specific Improved Technical Specifications, portions of some Specifications which are descriptive in nature regarding equipment, systems, actions, surveillances or programs are proposed to be relocated from the Specifications to the Bases, Updated Final Safety Analysis Report, procedures or other licensee controlled documents. The details associated with the involved specifications are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for compliance with the applicable specifications. Changes to the Bases are controlled in accordance with the proposed Bases Control Program described in Chapter 5 of the Improved Technical Specifications. Changes to the UFSAR and administrative procedures which control revisions to these relocated requirements are controlled in accordance with licensee controlled programs.

This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Less Restrictive-Generic" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes relocate requirements from the Technical Specifications to the Bases, Updated Final Safety Analysis Report, procedures or other licensee controlled documents. The documents containing the relocated requirements are subject to the change control of licensee controlled programs. Since any changes to these documents will be evaluated in accordance with the requirements of licensee controlled programs, no increase in the probability or consequences of an accident previously evaluated will be permitted without further NRC review. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. These changes do not introduce a new mode of plant operation. Since any future changes to these requirements will be evaluated in accordance with licensee controlled programs, the possibility of a new or different kind of accident from any accident previously evaluated will not be permitted without further NRC review. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed changes will not reduce a margin of safety because they do not impact any safety analysis assumptions. The requirements that are transposed from the Technical Specifications to other licensee controlled documents are the same as the existing Technical Specifications. Since any future changes to these requirements will be evaluated in accordance with the requirements of licensee controlled programs, no reduction in any margin of safety will be permitted without further NRC review. Therefore, these changes do not involve any reduction in a margin of safety.

LESS RESTRICTIVE SPECIFIC CHANGES
("L1" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change eliminates the Applicability of the RCS Pressure SL under conditions where the potential to challenge the SL is unlikely or not possible (with one or more reactor vessel closure bolts less than fully tensioned or with the head removed). With one or more reactor vessel head bolts less than fully tensioned, it is highly unlikely that the RCS can be pressurized greater than the SL pressure due to the low temperature over-pressure (LTOP) protection requirements. With the head removed, it is not possible to pressurize the RCS greater than the SL pressure. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change does eliminate the applicability of the RCS Pressure SL under conditions where the potential to challenge the SL is unlikely or not possible (with one or more reactor vessel closure bolts less than fully tensioned or with the head removed). The proposed change does not introduce a new mode of operation or changes in the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   There are no margins of safety related to any safety analysis that is dependent upon the proposed change. The requirements will continue to assure adequate protection from exceeding the RCS pressure SL. Therefore, this change does not involve a reduction in a margin of safety.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
NRC DOCKET NO. 50-261/LICENSE NO. DPR-23
REQUEST FOR TECHNICAL SPECIFICATION CHANGE
CONVERSION TO IMPROVED STANDARD TECHNICAL SPECIFICATIONS

ENVIRONMENTAL CONSIDERATIONS

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulator actions for categorical exclusion for performing an environmental assessment. A proposed change for an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; (3) result in an increase in individual or cumulative occupational radiation exposure. We have reviewed this request and determined that the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance with the amendment. The basis for this determination follows.

Proposed Change

This request proposes to change the technical specifications to be consistent with NUREG-1431; Standard Technical Specifications, Westinghouse Plants Revision 1, 04/07/95 within limitations imposed by plant specific design and licensing basis.

Basis

The proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons.

1. As demonstrated in the No Significant Hazards Evaluation, the proposed changes do not involve a significant hazards consideration.

2. These proposed changes are being made to establish consistency with the improved Standard Technical Specifications (ISTS) - Westinghouse Plants, NUREG 1431, Rev. 1, including approved generic changes, and do not involve physical changes to the facility, nor do they affect actual plant effluents.

3. These proposed changes are being made to establish consistency with the improved Standard Technical Specifications (ISTS) - Westinghouse Plants, NUREG 1431, Rev. 1, including approved generic changes and do not involve physical changes to the facility, and they do not significantly affect individual or cumulative occupational radiation exposures.

CTS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

[2.1.a] 2.1.1 Reactor Core SLs

cold leg (1)

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest ~~loop average~~ temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.

[2.2] 2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~X~~2735~~X~~ psig.

[6.7] 2.2 SL Violations

[M2] 2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

[M2] 2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

[M2] 2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.4 Within 24 hours, notify the [Plant Superintendent and Vice President – Nuclear Operations].

2.2.5 Within 30 days a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the [offsite review function], and the [Plant Superintendent, and Vice President – Nuclear Operations].

2.2.6 Operation of the unit shall not be resumed until authorized by the NRC.

TSTF-5

INSERT II2.1-1

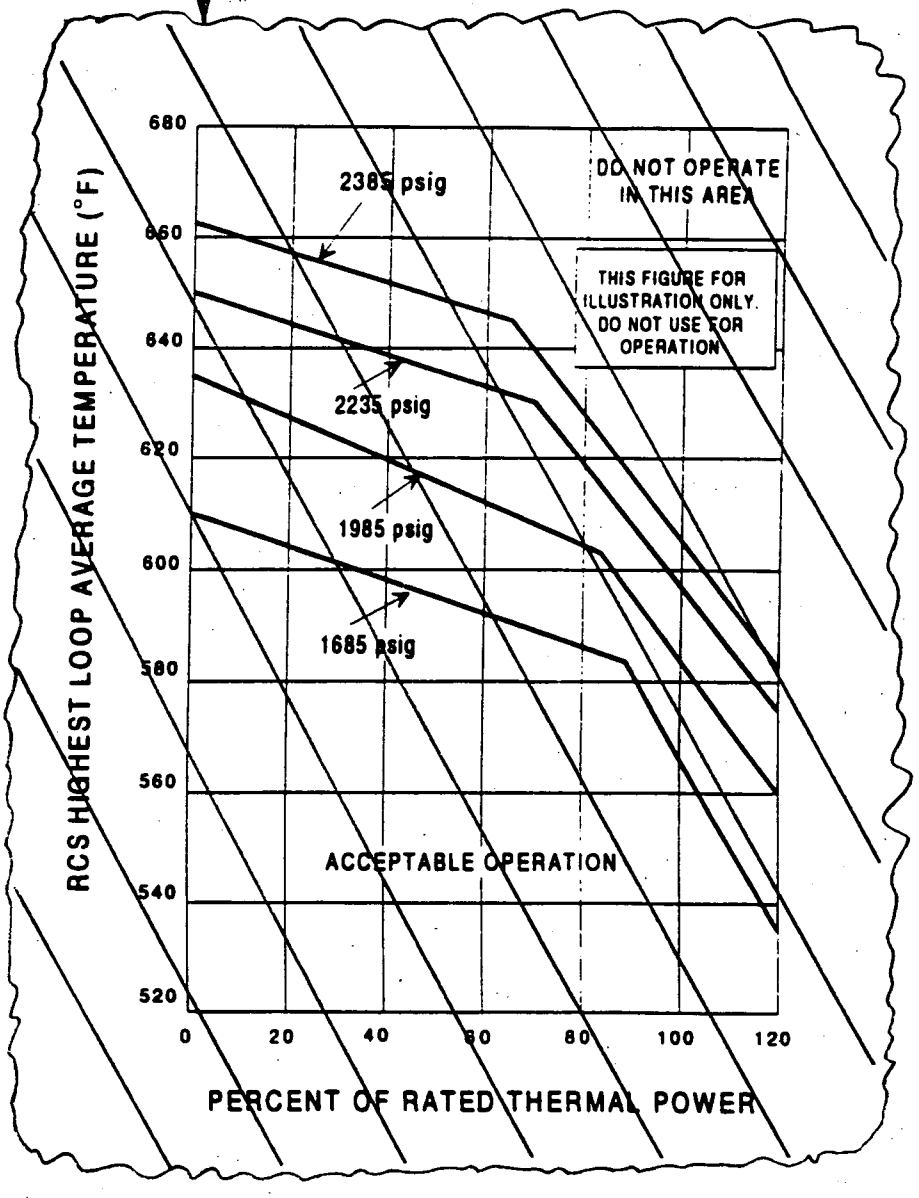
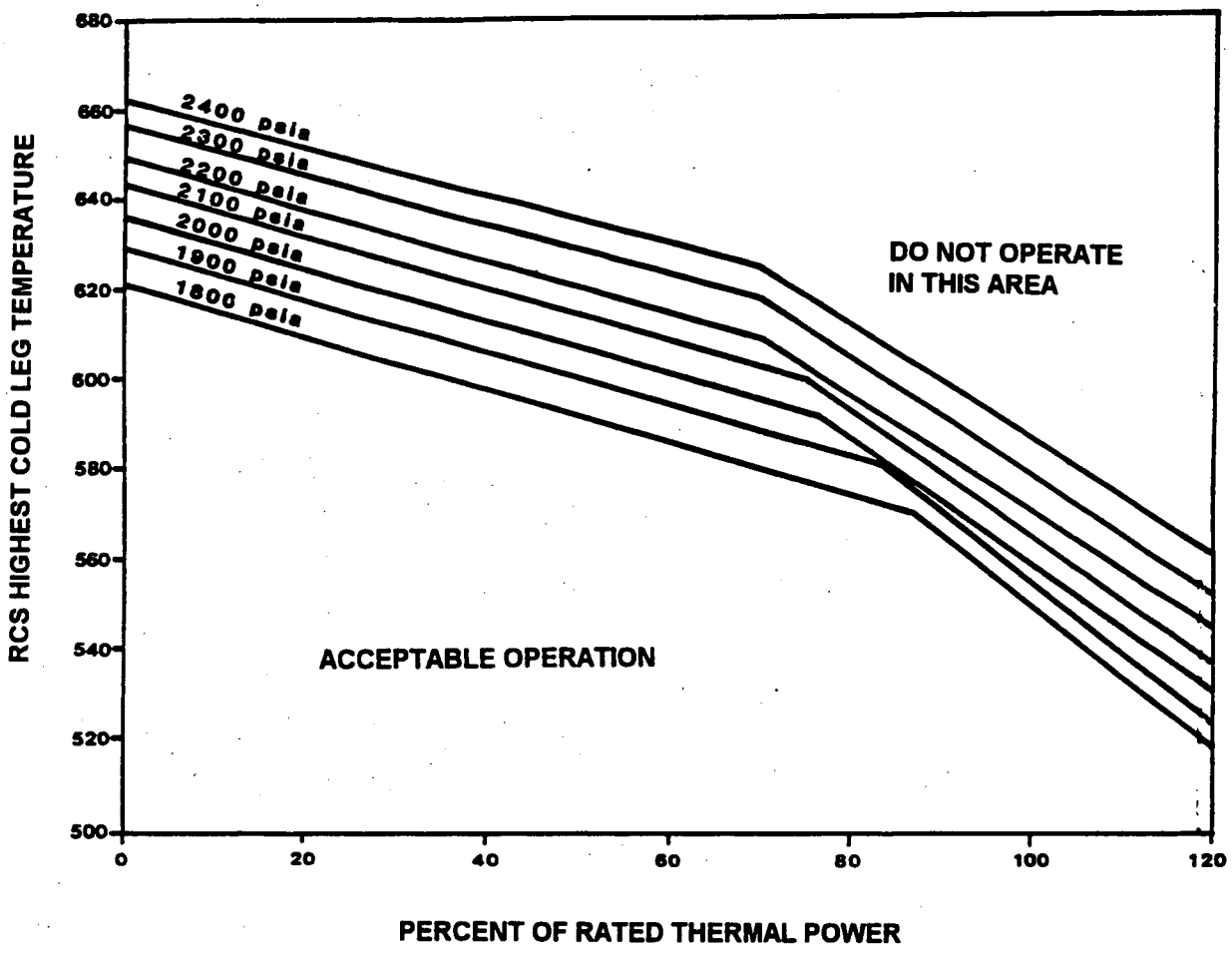


Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

INSERT 1



NOTE: BASED ON A MINIMUM RCS FLOW OF 97.3 x 10^6 lbm/hr

Insert II 2.1-1

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 2.0 - SAFETY LIMITS (SLs)

PART 4

MARKUP OF NUREG 1431, REVISION 1, "STANDARD TECHNICAL SPECIFICATIONS - WESTINGHOUSE PLANTS" (ISTS)

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 2.0 - SAFETY LIMITS (SLs)

PART 5

JUSTIFICATION FOR DIFFERENCES (JFDs) TO ISTS

1. ISTS 2.1.1 utilizes the "RCS highest loop average temperature" as one of the bounding parameters for the reactor core SLs. ITS 2.1.1 uses the "RCS highest cold leg temperature" to preserve consistency with the current licensing basis, specifically the limits specified in CTS figure 2.1-1. There is no material difference between the ITS SL curve expressed as a function of cold leg temperature and the ISTS curve expressed as a function of RCS average temperature. The limits specified in CTS figure 2.1-1 do not specifically state in which loop the RCS cold leg temperature measurement is made. The delineation of the use of the highest value for the three RCS cold leg temperatures in the ITS is conservative, since use of this value bounds the conditions in the remaining loops with respect to impact on the fuel SLs.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 2.0 - SAFETY LIMITS (SLs)

PART 6

MARKUP OF ISTS BASES

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

Insert B2.1.1-1

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

generation

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

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

(continued)

ITS Insert B2.1.1-1

The General Design Criteria (GDC) in existence at the time HBRSEP UNIT 2 was licensed for operation (July 1970) were contained in the proposed Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants," published in the Federal Register on July 11, 1967 (Ref. 1). Proposed GDC-6 required that the reactor core with its related controls and protection systems be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which had been stipulated and justified. The core and related auxiliary system designs provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the Updated Final Safety Analysis Report (UFSAR), Ref. 4) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The curves provided in Figure 2.1.1-1 show the loci of points of THERMAL POWER, RCS pressure, and reactor vessel inlet temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within the limits defined by the DNBR correlation. Figure 2.1.1-1 shows the allowable power level decreasing with increasing reactor vessel inlet temperature at selected pressurizer pressures for constant flow (i.e., three loop operation, minimum flow 97.3 x 10^6 lbm/hr). The area where clad integrity is assured is below these lines. The temperature limits at low power are considerably more conservative than would be required if they were based on the minimum allowable DNB ratio, but are set to preclude bulk boiling at the vessel exit. The safety limit curves given in Figure 2.1.1-1 are for constant flow conditions. These curves would not be applicable in cases where total reactor coolant flow is less than 97.3 x 10^6 lbm/hr. The evaluation of such an event would be based upon the analysis presented in Section 15.3 of the UFSAR.

The SL is higher than the limit calculated when the Axial Flux Difference (AFD) is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature and overpower ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Ref. 4).

(continued)

BASES (continued)

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The main steam safety valves and automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

If SL 2.1.1 is violated, the requirement to restore compliance and go to MODE 3 places the unit in a safe condition and in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Proposed Appendix A, 32FR10213, July 11, 1967.

2. XN-NF-711(P) Rev. 0, "XNB Addendum for 26 Inch Spacer."

3. ANF-1224(P) Rev.0, "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel."

4. UFSAR, Sections 3.1, 4.4, 7.2, and 15.0.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50 Proposed Appendix A (Ref. 1), GDC 9 "Reactor Coolant System Pressure Boundary" and GDC 34 "Reactor Coolant Pressure Boundary (RCPB) Rapid Propagation Failure Prevention," the reactor coolant pressure boundary design conditions are not to be exceeded during normal operations and transients. Also, in accordance with proposed GDC 33, "Reactor Coolant Pressure Boundary Capability," reactivity accidents, including rod ejection and inadvertent and sudden releases of energy to the coolant, do not result in damage to the RCPB.

The design pressure of the RCS is 2485 psig. During normal operation and transients, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components were hydrostatically tested at 3110 psig, according to the ASME Code requirements prior to initial operation with no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria."

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence safety valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the reactor is assumed to trip when the RCS pressure reaches the high RCS pressurizer pressure trip setpoint, the RCS pressurizer safety valves are assumed to open when the RCS pressure reaches the RCS safety valve setpoint, and the MSSVs on the secondary plant are assumed to open when the main steam pressure reaches MSSV settings.

The Reactor Protection System setpoints specified in Limiting Condition for Operations (LCO) 3.3.1, together with the settings of the RCS Pressurizer Safety Valves and MSSVs, provide pressure protection for normal operation and transients. The reactor high pressure trip setpoint specified in LCO 3.3.1 is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. Pressurizer power operated relief valves (PORVs);

b. Main steam power operated relief valves;

c. Steam Dump System;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray valves.

(continued)

BASES (continued)

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.1 (Ref. 5) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits.

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Proposed Appendix A, 32FR10213, July 11, 1967.

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

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

4. 10 CFR 100.

5. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 2.0 - SAFETY LIMITS (SLs)

PART 10

ISTS GENERIC CHANGES

Industry/TSTF Standard Technical Specification Change Traveler

Delete safety limit violation notification requirements

NUREGs Affected: ☒ 1430 ☒ 1431 ☒ 1432 ☒ 1433 ☒ 1434

Description:

Delete notification, reporting, and restart requirements if a safety limit is violated.

Justification:

This change deletes requirements from the Technical Specifications that are duplicative or contained in other regulations or required to comply with regulations (10 CFR 50.36). The change is also consistent with Revision 0 Change BWOG-09, which addressed several NRC and Industry initiatives to improve the content and presentation of Administrative Controls.

Affected Technical Specifications

2.1.1 Bases	Reactor Core Safety Limits	NUREG(s)- 1430 Only
	Change Description: Eliminate discussion of Actions 2.2.5, 2.2.6, 2.2.7, and 2.2.8, and References 3 and 4	
2.1.2 Bases	RCS Pressure Safety Limits	NUREG(s)- 1430 Only
	Change Description: Eliminate discussion of Actions 2.2.5, 2.2.6, 2.2.7, and 2.2.8, and References. 8 and 9	
2.2	Safety Limit Violations	NUREG(s)- 1430 Only
	Change Description: Eliminate Actions 2.2.5, 2.2.6, 2.2.7, 2.2.8	
2.1.1 Bases	Reactor Core Safety Limits	NUREG(s)- 1431 Only
	Change Description: Eliminate discussion of Actions 2.2.3, 2.2.4, 2.2.5, and 2.2.6, the label of 2.2.1, and References 5 and 6	
2.1.2 Bases	RCS Pressure Safety Limits	NUREG(s)- 1431 Only
	Change Description: Eliminate discussion of Actions 2.2.3, 2.2.4, 2.2.5, and 2.2.6, the labels of 2.2.2.1 and 2.2.2.1, and Refs. 7 and 8	
2.2	Safety Limit Violations	NUREG(s)- 1431 Only
	Change Description: Eliminate Actions 2.2.3, 2.2.4, 2.2.5, 2.2.6	
2.0	Safety Limit Violations (Analog and Digital)	NUREG(s)- 1432 Only
	Change Description: Eliminate Actions 2.2.3, 2.2.4, 2.2.5, 2.2.6	
2.1.1 Bases	Reactor Core Safety Limits (Analog)	NUREG(s)- 1432 Only
	Change Description: Eliminate discussion of Actions 2.2.3, 2.2.4, 2.2.5, and 2.2.6. Eliminate References 3 and 4.	
2.1.1 Bases	Reactor Core Safety Limits (Digital)	NUREG(s)- 1432 Only
	Change Description: Eliminate the label on 2.2.1. Eliminate discussion of Actions 2.2.3, 2.2.4, 2.2.5, and 2.2.6, and References 3 and 4	
2.1.2 Bases	RCS Pressure Safety Limits (Analog and Digital)	NUREG(s)- 1432 Only
	Change Description: Eliminate discussion of Actions 2.2.3, 2.2.4, 2.2.5, and 2.2.6, and References 7 and 8	
2.1.1 Bases	Reactor Core Safety Limits	NUREG(s)- 1433 Only
	Change Description: Eliminate 2.2.1, 2.2.3, 2.2.4, 2.2.5, the label on 2.2.2, and References 4 and 6	
2.2	Safety Limit Violations	NUREG(s)- 1433 Only
	Change Description: Revise 2.2.2. Eliminate 2.2.1, 2.2.3, 2.2.4, and 2.2.5.	
2.2.2 Bases	Reactor Coolant System Safety Limits	NUREG(s)- 1433 Only
	Change Description: Eliminate 2.2.1, 2.2.3, 2.2.4, 2.2.5, the label on 2.2.2, and References 7 and 8	

2.1.1 Bases	Reactor Core Safety Limits	NUREG(s)- 1434 Only
	Change Description: Eliminate 2.2.1, 2.2.3, 2.2.4, 2.2.5, the label on 2.2.2, and References 4 and 6	
2.2	Safety Limit Violations	NUREG(s)- 1434 Only
	Change Description: Revise2.2.2. Eliminate 2.2.1, 2.2.3, 2.2.4, and 2.2.5.	
2.2.2 Bases	Reactor Coolant System Safety Limits	NUREG(s)- 1434 Only
	Change Description: Eliminate 2.2.1, 2.2.3, 2.2.4, 2.2.5, the label on 2.2.2, and References 7 and 8	

WOG Review Information

WOG-2

Originating Plant: Date Provided to OG: 15-Mar-95 Needed By:

Owners Group History:

WOG-02, C.1

Owners Group Resolution: Approved Date: 11-Aug-95

TSTF Review Information

TSTF Received Date: 05-Sep-95 Date Distributed to OGs for Review: 05-Sep-95

OG Review Completed: ☒ BWOG ☒ WOG ☒ CEOG ☒ BWROG

TSTF History:

TSTF Resolution: Approved Date: 05-Sep-95 **TSTF- 5**

NRC Review Information

NRC Received Date: 05-Sep-95 NRC Reviewer: R. Tjader Reviewer Phone #:

Reviewer Comments:

10/4/95 - R. Tjader review complete, accept change.
10/4/95 to C. Grimes to review.
11/17/95 C. Grimes approved change.

Final Resolution: Approved Date: 27-Nov-95

Revision History

Revision 1 Revision Date: 08-Jan-96 Proposed by: TSTF

Revision Description:
In all NUREGs except the BWOG NUREG, all specification labels were deleted from the Safety Limit Violations Bases. There is no reason for the BWOG NUREG to be different. This was corrected.
In the CEOG NUREG (Digital) SL 2.1.1 Bases, References 3 and 4 were not deleted along with their occurrences in the Bases. This was corrected.
The Traveler stated that SL 2.2.1 is revised for the BWR NUREGs, when it was eliminated. This was corrected.

Resolution: Date:

Incorporation Into the NUREGs

File to BBS/LAN Date:

File to TSTF Date:

File Rev Incoporated:

File Rev Incorporated Date

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ [2735] psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.4 Within 24 hours, notify the [Plant Superintendent and Vice President—Nuclear Operations].

2.2.5 Within 30 days a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the [offsite review function], and the [Plant Superintendent, and Vice President—Nuclear Operations].

2.2.6 Operation of the unit shall not be resumed until authorized by the NRC.

BASES

APPLICABILITY (continued)

5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

~~2.2.1~~

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

~~2.2.3~~

~~If SL 2.1.1 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 5).~~

~~2.2.4~~

~~If SL 2.1.1 is violated, the Plant Superintendent and the Vice President—Nuclear Operations shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.~~

~~2.2.5~~

~~If SL 2.1.1 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 6). A copy of the report shall also be provided to the Plant Superintendent and the Vice President—Nuclear Operations.~~

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

~~2.2.6~~

~~If SL 2.1.1 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section [7.2].
3. WCAP-8746-A, March 1977.
4. WCAP-9273-NP-A, July 1985.
~~5. 10 CFR 50.72.~~
~~6. 10 CFR 50.73.~~

BASES (continued)

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS

~~The following SL violations are applicable to the RCS pressure SL.~~

~~2.2.2.1~~

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

~~2.2.2.2~~

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.3

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 7).

2.2.4

If the RCS pressure SL is violated, the Plant Superintendent and the Vice President–Nuclear Operations shall be notified within 24 hours. The 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.

2.2.5

If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the Plant Superintendent and the Vice President–Nuclear Operations.

2.2.6

If the RCS pressure SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.

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

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

4. 10 CFR 100.

(continued)

BASES

REFERENCES (continued)

5. FSAR, Section [7.2].

6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

~~7. 10 CFR 50.72.~~

~~8. 10 CFR 50.73.~~

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 2.0 - SAFETY LIMITS (SLs)

PART 8

PROPOSED HBRSEP, UNIT NO. 2 ITS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest cold leg temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

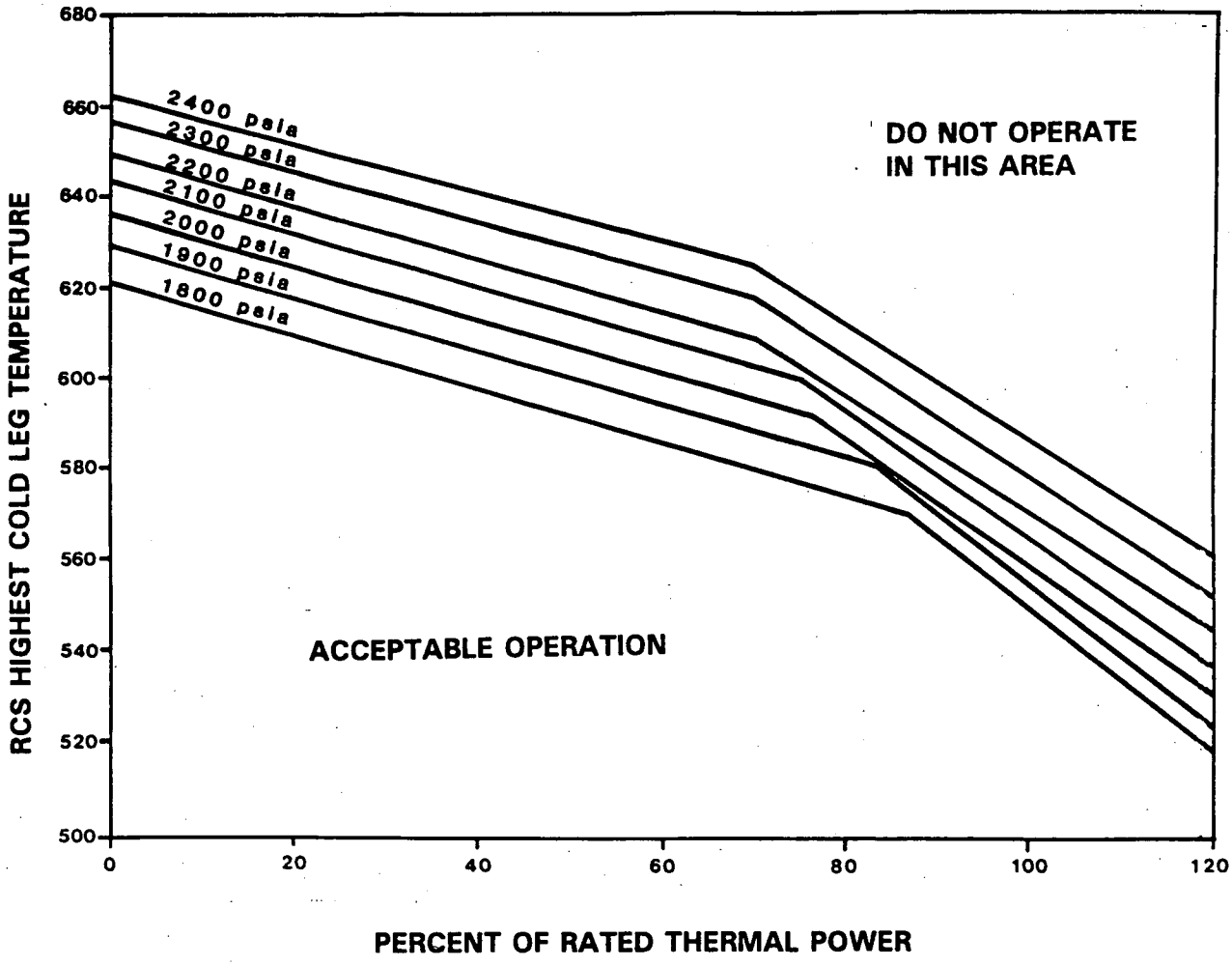
2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.



NOTE: BASED ON A MINIMUM RCS FLOW OF 97.3 x 10^6 lbm/hr

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety limits

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 2.0 - SAFETY LIMITS (SLs)

PART 9

PROPOSED BASES TO HBRSEP, UNIT NO. 2 ITS

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

The General Design Criteria (GDC) in existence at the time HBRSEP Unit No. 2 was licensed for operation (July 1970) were contained in the proposed Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants," published in the Federal Register on July 11, 1967 (Ref. 1). Proposed GDC-6 required that the reactor core with its related controls and protection systems be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which had been stipulated and justified. The core and related auxiliary system designs provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat generation rate (LHGR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHGR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

(continued)

BASES

BACKGROUND (continued)

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

The proper functioning of the Reactor Protection System (RPS) and main steam safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by maintaining the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB), and at this point there is a sharp reduction in the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters (i.e., thermal power, reactor coolant temperature and pressure) have been related to DNB through correlations. DNB correlations have been developed to

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum DNB ratio, or DNBR, during normal operational and anticipated transients, is restricted to the safety limit. A DNBR at the safety limit corresponds to a 95% probability, at a 95% confidence level, that DNB will not occur, and is chosen as an appropriate margin to DNB for all operating conditions. The DNBR safety limit is a conservative design value which is used as a basis for setting core safety limits. Based on rod bundle tests, no fuel damage is expected at this DNBR or greater. For the standard mixing vane fuel, the Siemens Power Corporation XNB correlation has a DNBR safety limit of 1.17 (Ref. 2) and for the high thermal performance fuel the Siemens ANFP correlation has a DNBR safety limit of 1.154 (Ref. 3).

The Reactor Trip System setpoints specified in Limiting Condition for Operations (LCO) 3.3.1, in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressurizer pressure, flow, core power distribution, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. Overtemperature ΔT trip;

b. Overpower ΔT trip;

c. Power Range Neutron Flux trip; and

d. Main steam safety valves.

Maintaining the DNBR above the limit ensures that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid and also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the Updated Final Safety Analysis Report (UFSAR), Ref. 4) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The curves provided in Figure 2.1.1-1 show the loci of points of THERMAL POWER, RCS pressure, and reactor vessel inlet temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within the limits defined by the DNBR correlation. Figure 2.1.1-1 shows the allowable power level decreasing with increasing reactor vessel inlet temperature at selected pressurizer pressures for constant flow (i.e., three loop operation, minimum flow 97.3 x 10^6 lbm/hr). The area where clad integrity is assured is below these lines. The temperature limits at low power are considerably more conservative than would be required if they were based on the minimum allowable DNB ratio, but are set to preclude bulk boiling at the vessel exit. The safety limit curves given in Figure 2.1.1-1 are for constant flow conditions. These curves would not be applicable in cases where total reactor coolant flow is less than 97.3 x 10^6 lbm/hr. The evaluation of such an event would be based upon the analysis presented in Section 15.3 of the UFSAR.

The SL is higher than the limit calculated when the Axial Flux Difference (AFD) is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature and overpower ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Ref. 4).

(continued)

BASES (continued)

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The main steam safety valves and automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

If SL 2.1.1 is violated, the requirement to restore compliance and go to MODE 3 places the unit in a safe condition and in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Proposed Appendix A, 32FR10213, July 11, 1967.

2. XN-NF-711(P) Rev. 0, "XNB Addendum for 26 Inch Spacer."

3. ANF-1224(P) Rev.0, "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel."

4. UFSAR, Sections 3.1, 4.4, 7.2, and 15.0.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50 Proposed Appendix A (Ref. 1), GDC 9 "Reactor Coolant System Pressure Boundary" and GDC 34 "Reactor Coolant Pressure Boundary (RCPB) Rapid Propagation Failure Prevention," the reactor coolant pressure boundary design conditions are not to be exceeded during normal operations and transients. Also, in accordance with proposed GDC 33, "Reactor Coolant Pressure Boundary Capability," reactivity accidents, including rod ejection and inadvertent and sudden releases of energy to the coolant, do not result in damage to the RCPB.

The design pressure of the RCS is 2485 psig. During normal operation and transients, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components were hydrostatically tested at 3110 psig, according to the ASME Code requirements prior to initial operation with no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria."

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence safety valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the reactor is assumed to trip when the RCS pressure reaches the high RCS pressurizer pressure trip setpoint, the RCS pressurizer safety valves are assumed to open when the RCS pressure reaches the RCS safety valve setpoint, and the MSSVs on the secondary plant are assumed to open when the main steam pressure reaches MSSV settings.

The Reactor Protection System setpoints specified in Limiting Condition for Operations (LCO) 3.3.1, together with the settings of the RCS Pressurizer Safety Valves and MSSVs, provide pressure protection for normal operation and transients. The reactor high pressure trip setpoint specified in LCO 3.3.1 is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. Pressurizer power operated relief valves (PORVs);

b. Main steam power operated relief valves;

c. Steam Dump System;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray valves.

(continued)

BASES (continued)

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.1 (Ref. 5) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits.

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Proposed Appendix A, 32FR10213, July 11, 1967.

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

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

4. 10 CFR 100.

5. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 2.0 - SAFETY LIMITS (SLs)

PART 10

ISTS GENERIC CHANGES

Industry/TSTF Standard Technical Specification Change Traveler

Delete safety limit violation notification requirements

NUREGs Affected: ☒ 1430 ☒ 1431 ☒ 1432 ☒ 1433 ☒ 1434

Description:

Delete notification, reporting, and restart requirements if a safety limit is violated.

Justification:

This change deletes requirements from the Technical Specifications that are duplicative or contained in other regulations or required to comply with regulations (10 CFR 50.36). The change is also consistent with Revision 0 Change BWOG-09, which addressed several NRC and Industry initiatives to improve the content and presentation of Administrative Controls.

Affected Technical Specifications

2.1.1 Bases	Reactor Core Safety Limits	NUREG(s)- 1430 Only
	Change Description: Eliminate discussion of Actions 2.2.5, 2.2.6, 2.2.7, and 2.2.8, and References 3 and 4	
2.1.2 Bases	RCS Pressure Safety Limits	NUREG(s)- 1430 Only
	Change Description: Eliminate discussion of Actions 2.2.5, 2.2.6, 2.2.7, and 2.2.8, and References. 8 and 9	
2.2	Safety Limit Violations	NUREG(s)- 1430 Only
	Change Description: Eliminate Actions 2.2.5, 2.2.6, 2.2.7, 2.2.8	
2.1.1 Bases	Reactor Core Safety Limits	NUREG(s)- 1431 Only
	Change Description: Eliminate discussion of Actions 2.2.3, 2.2.4, 2.2.5, and 2.2.6, the label of 2.2.1, and References 5 and 6	
2.1.2 Bases	RCS Pressure Safety Limits	NUREG(s)- 1431 Only
	Change Description: Eliminate discussion of Actions 2.2.3, 2.2.4, 2.2.5, and 2.2.6, the labels of 2.2.2.1 and 2.2.2.1, and Refs. 7 and 8	
2.2	Safety Limit Violations	NUREG(s)- 1431 Only
	Change Description: Eliminate Actions 2.2.3, 2.2.4, 2.2.5, 2.2.6	
2.0	Safety Limit Violations (Analog and Digital)	NUREG(s)- 1432 Only
	Change Description: Eliminate Actions 2.2.3, 2.2.4, 2.2.5, 2.2.6	
2.1.1 Bases	Reactor Core Safety Limits (Analog)	NUREG(s)- 1432 Only
	Change Description: Eliminate discussion of Actions 2.2.3, 2.2.4, 2.2.5, and 2.2.6. Eliminate References 3 and 4.	
2.1.1 Bases	Reactor Core Safety Limits (Digital)	NUREG(s)- 1432 Only
	Change Description: Eliminate the label on 2.2.1. Eliminate discussion of Actions 2.2.3, 2.2.4, 2.2.5, and 2.2.6, and References 3 and 4	
2.1.2 Bases	RCS Pressure Safety Limits (Analog and Digital)	NUREG(s)- 1432 Only
	Change Description: Eliminate discussion of Actions 2.2.3, 2.2.4, 2.2.5, and 2.2.6, and References 7 and 8	
2.1.1 Bases	Reactor Core Safety Limits	NUREG(s)- 1433 Only
	Change Description: Eliminate 2.2.1, 2.2.3, 2.2.4, 2.2.5, the label on 2.2.2, and References 4 and 6	
2.2	Safety Limit Violations	NUREG(s)- 1433 Only
	Change Description: Revise 2.2.2. Eliminate 2.2.1, 2.2.3, 2.2.4, and 2.2.5.	
2.2.2 Bases	Reactor Coolant System Safety Limits	NUREG(s)- 1433 Only
	Change Description: Eliminate 2.2.1, 2.2.3, 2.2.4, 2.2.5, the label on 2.2.2, and References 7 and 8	

2.1.1 Bases	Reactor Core Safety Limits	NUREG(s)- 1434 Only
	Change Description:	Eliminate 2.2.1, 2.2.3, 2.2.4, 2.2.5, the label on 2.2.2, and References 4 and 6
2.2	Safety Limit Violations	NUREG(s)- 1434 Only
	Change Description:	Revise2.2.2. Eliminate 2.2.1, 2.2.3, 2.2.4, and 2.2.5.
2.2.2 Bases	Reactor Coolant System Safety Limits	NUREG(s)- 1434 Only
	Change Description:	Eliminate 2.2.1, 2.2.3, 2.2.4, 2.2.5, the label on 2.2.2, and References 7 and 8

WOG Review Information

WOG-2

Originating Plant: Date Provided to OG: 15-Mar-95 Needed By:

Owners Group History:

WOG-02, C.1

Owners Group Resolution: Approved Date: 11-Aug-95

TSTF Review Information

TSTF Received Date: 05-Sep-95 Date Distributed to OGs for Review: 05-Sep-95

OG Review Completed: ☒ BWOG ☒ WOG ☒ CEOG ☒ BWROG

TSTF History:

TSTF Resolution: Approved Date: 05-Sep-95 **TSTF- 5**

NRC Review Information

NRC Received Date: 05-Sep-95 NRC Reviewer: R. Tjader Reviewer Phone #:

Reviewer Comments:

10/4/95 - R. Tjader review complete, accept change.
10/4/95 to C. Grimes to review.
11/17/95 C. Grimes approved change.

Final Resolution: Approved Date: 27-Nov-95

Revision History

Revision 1 Revision Date: 08-Jan-96 Proposed by: TSTF

Revision Description:
In all NUREGs except the BWOG NUREG, all specification labels were deleted from the Safety Limit Violations Bases. There is no reason for the BWOG NUREG to be different. This was corrected.
In the CEOG NUREG (Digital) SL 2.1.1 Bases, References 3 and 4 were not deleted along with their occurrences in the Bases. This was corrected.
The Traveler stated that SL 2.2.1 is revised for the BWR NUREGs, when it was eliminated. This was corrected.

Resolution: Date:

Incorporation Into the NUREGs

File to BBS/LAN Date:

File to TSTF Date:

File Rev Incoporated:

File Rev Incorporated Date

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ [2735] psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.4 Within 24 hours, notify the [Plant Superintendent and Vice President—Nuclear Operations].

2.2.5 Within 30 days a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the [offsite review function], and the [Plant Superintendent, and Vice President—Nuclear Operations].

2.2.6 Operation of the unit shall not be resumed until authorized by the NRC.

BASES

APPLICABILITY (continued)

5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

~~2.2.1~~

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

~~2.2.3~~

~~If SL 2.1.1 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 5).~~

~~2.2.4~~

~~If SL 2.1.1 is violated, the Plant Superintendent and the Vice President—Nuclear Operations shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.~~

~~2.2.5~~

~~If SL 2.1.1 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 6). A copy of the report shall also be provided to the Plant Superintendent and the Vice President—Nuclear Operations.~~

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

~~2.2.6~~

~~If SL 2.1.1 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section [7.2].
3. WCAP-8746-A, March 1977.
4. WCAP-9273-NP-A, July 1985.
5. ~~10 CFR 50.72~~.
6. ~~10 CFR 50.73~~.

BASES (continued)

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS

~~The following SL violations are applicable to the RCS pressure SL.~~

~~2.2.2.1~~

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

~~2.2.2.2~~

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.3

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 7).

2.2.4

If the RCS pressure SL is violated, the Plant Superintendent and the Vice President–Nuclear Operations shall be notified within 24 hours. The 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.

2.2.5

If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the Plant Superintendent and the Vice President–Nuclear Operations.

2.2.6

If the RCS pressure SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.

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

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

4. 10 CFR 100.

(continued)

BASES

REFERENCES (continued)

5. FSAR, Section [7.2].

6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

~~7. 10 CFR 50.72.~~

~~8. 10 CFR 50.73.~~

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
CONVERSION PACKAGE SECTION 3.0

ITS CONVERSION PACKAGE

CHAPTER 3.0 - LCO AND SR APPLICABILITY

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.0 - LCO AND SR APPLICABILITY

PART 1

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

Add LCO 3.0.1 (A2)

Add LCO 3.0.2 (A3)

(A1)

[3.0] 3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY

[LCO 3.0.3] Except as otherwise provided for in each specification, if a Limiting Condition for Operation cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in hot shutdown within eight hours and in COLD SHUTDOWN within the next 30 hours unless corrective measures are taken that permit operation under the permissible Limiting Condition for Operation statements for the specified time interval as measured from initial discovery or until the reactor is placed in a condition in which the specification is not applicable. (A4) (M1) (A4)

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those Reactor Coolant System conditions which must be met to assure safe reactor operation.

Specification

3.1.1 Operational Components

3.1.1.1 Coolant Pumps

a. With reactor power less than 2% of rated thermal power and less than two reactor coolant pumps in operation, one of the following actions shall be taken:

1. maintain a shutdown margin of at least 4% Δk/k, or
2. open the lift disconnect switches for all control rods not fully withdrawn, or
3. open reactor trip breakers.

See 3.4.5

Add LCO 3.0.4 (M2)

Add LCO 3.0.5 (L1)

Add LCO 3.0.6 (A6)

Add LCO 3.0.7 (A7)

3.3.7 Extended Maintenance

When it is determined that maintenance to restore components or systems to an operable condition will last longer than periods specified, the circumstances of the extended maintenance and the estimated date for returning the components or systems to an operable condition shall be promptly reported to the Director - Office of Nuclear Reactor Regulation and to the Director - Region II Office of Inspection and Enforcement. The purpose of prompt reporting is to allow the NRC to review the circumstances of the request for extended outage and to render a timely decision on whether to extend the specified out-of-service period while reactor operations continue.

A12

Basis

During low temperature physics tests, there is a negligible amount of stored energy in the reactor coolant, therefore an accident comparable in severity to a Design Basis Accident is not possible, and the engineering safety features systems are not required.

The operable status of the various systems and components is to be demonstrated by periodic tests, defined by Specification 4.5. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the system design and thereby limits the ability to tolerate additional equipment failures. For this reason, the unit is allowed to operate only for a limited time as specified when this condition occurs.

A11

Specification 3.0

A1

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

Specification

4.1.1 Calibration, testing, and checking of instrumentation channels shall be performed as specified in Table 4.1-1.

4.1.2 Sampling tests shall be conducted as specified in Table 4.1-2.

4.1.3 Equipment tests shall be conducted as specified in Table 4.1-3.

A10

Basis

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action, and a check supplements this type of built-in surveillance.

A11

Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequencies set forth are deemed adequate for reactor and steam system instrumentation.

Specification 3.0

For frequencies specified as "once" the above interval extension does not apply

If a completion time requires periodic performance on a "once per..." basis, the Frequency Extension applies to each performance after the initial performance

A1 M3 L2

3

4.0 SURVEILLANCE REQUIREMENTS

A9

[SR 3.0.2] Specified intervals may be adjusted plus ~~or minus~~ 25% to accommodate normal test schedules.

A8

[SR 3.0.1] Performance of any surveillance test outlined in these specifications is not required when the system or component is out of service as permitted by the Limiting Conditions for Operation. Prior to returning the system to service, the specified calibration and testing surveillance shall be performed.

LA1

[SR 3.0.3] If it is discovered that a Surveillance Requirement, as defined by Specification 4.0 and 4.0.1(e), was not performed within its specified frequency, then compliance with the requirement to declare that the Technical Specifications requirements are not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

LCO

A8

If the Surveillance is not performed within the delay period, then the Technical Specifications requirements must immediately be declared not met, and the applicable action requirements must be undertaken.

LCO

Conditions(s) entered

When the Surveillance is performed within the delay period and the Surveillance is not met, the Technical Specifications requirements must be immediately declared not met and the applicable action requirements must be undertaken.

LCO

M2

Add SR 3.0.4

Conditions(s) entered

4.0.1 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

see 5.0

c. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

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PART 2

DISCUSSION OF CHANGES (DOCS) FOR CTS MARKUP

ADMINISTRATIVE CHANGES

A1 In the conversion of the H. B. Robinson Steam Electric Plant (HBRSEP), Unit 2 Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in the Standard Technical Specifications, Westinghouse Plants, NUREG 1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS Specification 3.0 is revised to adopt ISTS Specification LCO 3.0.1 in the ITS. This Specification provides clarity with regard to when LCOs must be met, and where any exceptions can be found. This change is therefore administrative, and has no adverse impact on safety.

A3 CTS Specification 3.0 is revised to adopt ISTS Specification LCO 3.0.2 in the ITS. This Specification provides clarity with regard to the actions required to be taken upon discovery of a failure to meet an LCO. This change is therefore administrative, and has no adverse impact on safety.

A4 CTS Specification 3.0 is revised to adopt ISTS Specification LCO 3.0.3 text in the ITS:

a. The CTS phrase, "Except as otherwise provided for in each specification," is replaced with the ISTS phrase, "Exceptions to this Specification are stated in the individual Specifications," to clarify where exceptions to this LCO can be found.

b. The CTS phrase, "if a Limiting Condition for Operation cannot be satisfied because of circumstances in excess of those addressed in the specification," is replaced with the ISTS phrase, When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS," to specifically state the circumstances which require compliance with this LCO.

c. The CTS phrase, "unless corrective measures are taken that permit operation under the permissible Limiting Condition for Operation statements for the specified time interval as measured from initial discovery or until the reactor is placed in a condition in which the specification is not applicable," is replaced with the ISTS phrase "Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the

actions required by LCO 3.0.3 is not required," to clarify ambiguities regarding the termination of actions related to this LCO.

This change is therefore administrative, and has no adverse impact on safety.

A5 Not used.

A6 CTS Specification 3.0 is revised to adopt ISTS Specification LCO 3.0.6 in the ITS. This Specification provides guidance related to the appropriate actions to be taken when an inoperability of a support system also results in the inoperability of one or more related supported system(s). This Specification removes inconsistencies and ambiguities associated with inoperabilities in support and supported systems. The ISTS was developed to include LCO 3.0.6, and a new program, Specification 5.5.15, Safety Function Determination Program, to resolve the application of LCOs to support and supported systems. This change is therefore administrative, and has no adverse impact on safety.

A7 CTS Specification 3.0 is revised to adopt ISTS Specification LCO 3.0.7 in the ITS. This Specification provides guidance with regard to meeting Test Exceptions LCOs in ITS Specification 3.1.9, which allows certain Technical Specification requirements to be changed (i.e., made applicable in part or whole, or suspended) to permit performance of special tests or operations which otherwise could not be performed. This Specification eliminates confusion which would otherwise exist as to which LCOs apply during performance of a special test or operation. This change is therefore administrative, and has no adverse impact on safety.

A8 CTS Specification 4.0 is revised to adopt the ISTS Specification SR 3.0.1 text in the ITS. The CTS sentence, "Performance of any surveillance test outlined in these specifications is not required when the system or component is out of service as permitted by the Limiting Conditions for Operation," is replaced with the ISTS sentence, "Surveillances do not have to be performed on inoperable equipment or variables outside specified limits." Although not explicitly stated, the complementary requirement that Surveillances must be performed on equipment required to be OPERABLE (in accordance with applicable Technical Specification requirements) can be inferred. As such, the CTS requirements are consistent with the ITS.

ISTS Specification SR 3.0.1 also clarifies that failure to meet a Surveillance means failure to meet the LCO, and that such failure can be experienced between performances, as well as during performances of the Surveillance. This is consistent with CTS Surveillance Requirements when applied in conjunction with the CTS definition of OPERABLE.

Upon discovery a Surveillance Requirement was not performed within its specified frequency, CTS 4.0 permits an extension up 24 hours or the limits of the applicable Frequency, whichever is less. CTS 4.0 requires, if the SR is not met or not performed during the delay period, the Technical Specification requirements must be immediately declared not met and applicable actions taken. Although not explicitly stated, it can be inferred, that if the delay period is not invoked the Technical Specification requirements must be immediately declared not met and applicable actions taken., This is consistent with the requirement of ITS SR 3.0.1 and SR 3.0.3.

A9 CTS Specification 4.0 is revised to adopt the ISTS Specification SR 3.0.2 text in the ITS. The CTS sentence, "Specified intervals may be adjusted plus or minus 25% to accommodate normal test schedules," is replaced with the ISTS sentence, "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met," for clarification and to establish what constitutes meeting the specified Frequency of each SR. The CTS provision to adjust the specified interval by minus 25% is not retained. Elimination of this provision is considered administrative and is consistent with ISTS, since performance within the shortened interval is still within original interval.

The ISTS sentence, "Exceptions to this Specification are stated in the individual Specifications," is added to acknowledge the explicit use of exceptions in various Surveillances. This change is therefore administrative, and has no adverse impact on safety.

A10 CTS 4.1.1 explicitly requires calibration, testing and checking of instrument channels be performed as specified in Table 4.1-1. CTS 4.1.2 requires sampling tests be conducted as specified in Table 4.1-2. CTS 4.1.3 requires equipment tests be conducted as specified in Table 4.1-3. These specifications require the performance of the surveillances as specified in the individual tables but are not unique to a particular surveillance requirement. These specifications overlap other similar requirements specified in CTS 4.0 and are not separately retained in the ITS. Performance of SRs are required by ITS SR 3.0.1 consistent with the Applicabilities for the individual specifications. Therefore, elimination of these CTS specifications is considered administrative and is consistent with ISTS.

A11 The CTS Bases are not retained in the ITS, but are replaced in their entirety. The ITS includes significantly expanded and improved Bases. The Bases do not define or impose any specific requirements but serve to explain, clarify and document the reasons (i.e., Bases) for the associated specification. The Bases are not part of the Technical

Specifications required by 10 CFR 50.36. Therefore, this is an administrative change.

A12 CTS 3.3.7 provides Administrative Requirements to notify the NRC when maintenance to restore components or systems will exceed the periods specified. The requirements of this specification were rendered moot when CTS 3.0 was adopted in amendment 67. When an LCO cannot be met because of circumstances in excess of those addressed in the specification, CTS 3.0 requires the unit be placed in Hot Shutdown within 8 hours and Cold Shutdown within an additional 38 hours. Since the requirements of CTS 3.3.7 are obviated by the more restrictive requirements of CTS 3.0, the deletion of CTS 3.3.7 is considered an administrative change and is consistent with ISTS.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS Specification 3.0 is revised to adopt ISTS Specification LCO 3.0.3 text in the ITS. The CTS requires that, if an LCO cannot be met and there is no specific action required to be taken, the unit be placed in Hot Shutdown within 8 hours and in Cold Shutdown within the next 30 hours. The ISTS requires that, if an LCO cannot be met and there is no specific action required to be taken, the unit be placed in Hot Standby (MODE 3) within 7 hours, Hot Shutdown (MODE 4) within 13 hours, and Cold Shutdown (MODE 5) within 37 hours. The ISTS MODE 3 specification of 7 hours imposes a more restrictive requirement by one hour. An additional restraint imposed that is not specified in the CTS, is that the unit be in MODE 4 within 13 hours. The time allowed to achieve cold shutdown in the CTS is 38 hours, and the time allowed in the ISTS to achieve cold shutdown is 37 hours, resulting in the ISTS being more restrictive by one hour. This change imposes more restrictive requirements, and therefore has no adverse impact on safety.

M2 CTS Specifications 3.0 and 4.0 are revised to adopt ISTS Specifications LCO 3.0.4 and SR 3.0.4 in the ITS. These Specifications provide guidance related to MODE and operating condition entry when an LCO is not met. They also clarify those MODE changes permitted when required to comply with ACTIONS. The CTS does not preclude entry into a MODE in which compliance with a Specification applicable to that MODE is not met at the time of entry. This change imposes more restrictive requirements, and therefore has no adverse impact on safety.

M3 The statement, "For Frequencies specified as "once," the above interval extension does not apply," is added to clarify that the 1.25 times the interval specified in the Frequency does not apply to certain Surveillances. This is because the interval extension concept is based on scheduling flexibility for repetitive performances, and these Surveillances are not repetitive in nature, and essentially have no

"interval...as measured from the previous performance." This precludes the ability to extend these performances, and is therefore an additional restriction. The current Specification can be seen to allow the extension to apply to all Surveillances.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS Specification 4.0 is revised to adopt ISTS Specification SR 3.0.2 in the ITS. The CTS sentence, "Prior to returning the system to service, the specified calibration and testing surveillance shall be performed," is replaced with the ISTS sentence, "Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status," and relocated to the Bases for SR 3.0.1. This detail is not required to be in the ITS to provide adequate protection of the health and safety of the public, since it provides details of a clarification nature, which are not pertinent to the actual surveillance requirement, but rather describe acceptable methods of compliance, and more appropriately belong in the Bases. Since these details are not necessary to adequately describe actual surveillance requirements, they can be relocated to the Bases with no adverse impact on safety. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Changes to the Bases are controlled in accordance with the provisions of 10 CFR 50.59. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. This change is consistent with NUREG-1431. Therefore, relocation of this detail is acceptable.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 LCO 3.0.5 is added to provide an exception to LCO 3.0.2 for instances where restoration of inoperable equipment to an OPERABLE status could not be performed while continuing to comply with Required Actions. Many Technical Specification ACTIONS require an inoperable component to be removed from service, such as: maintaining an isolation valve closed or tripping an inoperable instrument channel. To allow the performance of Surveillance Requirements to demonstrate the OPERABILITY of the equipment being returned to service, or to demonstrate the OPERABILITY of other equipment or variables within limits, which otherwise could not be performed without returning the equipment to service, an exception to these Required Actions is necessary. LCO 3.0.5 is necessary to establish an allowance that, although informally utilized in restoration

of inoperable equipment, is not formally recognized in the current TS. Without this allowance certain components could not be restored to OPERABLE status and a plant shutdown would ensue. Clearly, it is not the intent or desire that the Technical Specifications preclude the return to service of a suspected OPERABLE component to confirm its OPERABILITY. This allowance is deemed to represent a more stable, safe operation than requiring a plant shutdown to complete the restoration and confirmatory testing.

L2 The statement "If a Completion Time requires periodic performance on a "once per..." basis, the above Frequency extension applies to each performance after the initial performance," is added to allow the 1.25 times the interval specified in the Frequency concept to apply to periodic Required Actions. This provides the consistency in scheduling flexibility for all performances of periodic requirements, whether they are Surveillances or Required Actions. The intent remains to perform the activity, on the average, once during each specified interval.

TECHNICAL CHANGES - LESS RESTRICTIVE (RELOCATION)

None

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NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC), AND BASIS FOR CATEGORICAL EXCLUSION FROM 10 CFR 51.22

ADMINISTRATIVE CHANGES
("A" Labeled Comments/Discussions)

In the conversion of the HBRSEP Unit No. 2 Technical Specifications to the proposed plant specific Improved Technical Specifications certain wording preferences or conventions are being adopted which do not result in technical changes (either actual or interpretational). Editorial changes, clarification, reformatting, rewording and revised numbering are being adopted to make the improved Technical Specifications consistent with NUREG 1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Administrative" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes consist of editorial changes and clarification, reformatting, rewording and renumbering of the current Technical Specifications. This process does not involve any technical changes to existing requirements. As such, these changes are administrative in nature and do not impact initiators of analyzed events or alter any assumptions relative to mitigation of accident or transient events. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. The proposed changes do not impose or eliminate any requirements. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed changes will not reduce a margin of safety because they do not impact any safety analysis assumptions. These changes are administrative in nature and, as such, do not impact any technical requirements. Therefore, these changes do not involve any reduction in a margin of safety.

MORE RESTRICTIVE CHANGES
("M" Labeled Comments/Discussions)

The HBRSEP Unit No. 2 Technical Specifications are proposed to be modified in some areas to impose more restrictive requirements than currently exist. These more restrictive changes are being imposed to be consistent with NUREG 1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "More Restrictive" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes provide requirements determined to be more restrictive than the current Technical Specifications requirements for operation of the facility. These more restrictive requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. These changes have been confirmed to ensure that no previously evaluated accident has been adversely affected. The more restrictive requirements being proposed enhance assurance that process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis of the unit. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. These changes do impose new or additional requirements which are consistent with assumptions made in the safety analysis and licensing basis. The additional requirements include new Surveillance Requirements, more restrictive Frequencies and Completion Times, new LCOs, more restrictive Required Actions and Applicabilities, and other operational restrictions that enhance safe operation. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact or increases the margin of plant safety. Each of the changes in this

category, while providing new or additional requirements designed to enhance plant safety, is consistent with the safety analyses and licensing basis. Therefore, these changes do not involve a reduction in a margin of safety.

LESS RESTRICTIVE-GENERIC CHANGES
("LA" Labeled Comments/Discussions)

In the conversion of the HBRSEP Unit No. 2 Technical Specifications to the proposed plant specific Improved Technical Specifications, portions of some Specifications which are descriptive in nature regarding equipment, systems, actions, surveillances or programs are proposed to be relocated from the Specifications to the Bases, Updated Final Safety Analysis Report, procedures or other licensee controlled documents. The details associated with the involved specifications are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for compliance with the applicable specifications. Changes to the Bases are controlled in accordance with the proposed Bases Control Program described in Chapter 5 of the Improved Technical Specifications. Changes to the UFSAR and administrative procedures which control revisions to these relocated requirements are controlled in accordance with licensee controlled programs.

This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Less Restrictive-Generic" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes relocate requirements from the Technical Specifications to the Bases, Updated Final Safety Analysis Report, procedures or other licensee controlled documents. The documents containing the relocated requirements are subject to the change control of licensee controlled programs. Since any changes to these documents will be evaluated in accordance with the requirements of licensee controlled programs, no increase in the probability or consequences of an accident previously evaluated will be permitted without further NRC review. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. These changes do not introduce a new mode of plant operation. Since any future changes to these requirements will be evaluated in accordance with licensee controlled programs, the possibility of a new or different kind of accident from any accident previously evaluated will not be permitted without further NRC review. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed changes will not reduce a margin of safety because they do not impact any safety analysis assumptions. The requirements that are transposed from the Technical Specifications to other licensee controlled documents are the same as the existing Technical Specifications. Since any future changes to these requirements will be evaluated in accordance with the requirements of licensee controlled programs, no reduction in any margin of safety will be permitted without further NRC review. Therefore, these changes do not involve any reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("Lx" Labeled Comments/Discussions)

L1 CHANGE

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The addition of LCO 3.0.5 allows restoration of equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. Temporarily returning inoperable equipment to service may in some cases increase the probability of a previously evaluated accident. However, the potential impact of temporarily returning the equipment to service is considered to be insignificant since the equipment will be restored to a condition which is expected to provide the required safety function. As stated in Generic Letter 87-09, "The vast majority of surveillances do in fact demonstrate that systems or components are operable." Also, returning the equipment to service under administrative controls will promote timely restoration

of the operability of the equipment and reduce the probability of any events that may have been prevented by such operable equipment. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated.

Since the equipment to be restored is already out of service, the availability of the equipment has been previously considered in the evaluation of consequences of an accident. Temporarily returning the equipment to service in a state which is expected to function as required to mitigate the consequences of a previously analyzed accident will promote timely restoration of the operability of the equipment and restore the capabilities of the equipment to mitigate the consequences of any events as previously analyzed. Therefore, the change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Operation with the inoperable equipment temporarily restored to service is not considered a new mode of operation since existing procedures and administrative controls prevent the restoration of equipment to service until it is considered capable of providing the required safety functions.

Performance of the surveillance is considered to be a confirmatory check of that capability which demonstrates that the equipment is indeed operable in the majority of the cases. For those times when equipment which may be temporarily returned to service under administrative controls is subsequently determined to be inoperable, the resulting condition is comparable to the equipment having been determined to be inoperable during operation, with continued operation for a specified time allowed to complete required actions. Since this condition has been previously evaluated in the development of the current Technical Specifications, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

Temporarily returning inoperable equipment to service for the purpose of confirming operability places the plant in a condition which has been previously evaluated and determined to be acceptable for short periods. Additionally, the equipment has been determined to be in a condition which provides the previously determined margin of safety. The performance of the surveillance simply confirms the expected result and capability of the equipment. Therefore, the change does not involve a significant reduction in a margin of safety.

L2 Change

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The application of the 25% extension to Required Action Completion Times which have a specified frequency on a periodic "once per" basis has been determined to not significantly degrade the reliability that results from performing the surveillance at a specified frequency. As stated in Generic Letter 87-09, "The vast majority of surveillances do in fact demonstrate that systems or components are operable." Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

3. Does this change involve a significant reduction in a margin of safety?

   The application of the 25% extension to Required Action Completion Times which have a specified frequency on a periodic "once per" basis has been determined to not significantly degrade the reliability that results from performing the surveillance at a specified frequency. As stated in Generic Letter 87-09, "The vast majority of surveillances do in fact demonstrate that systems or components are operable." Therefore, the proposed change does not involve a significant reduction in the margin of safety.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
NRC DOCKET NO. 50-261/LICENSE NO. DPR-23
REQUEST FOR TECHNICAL SPECIFICATION CHANGE
CONVERSION TO IMPROVED STANDARD TECHNICAL SPECIFICATIONS

ENVIRONMENTAL CONSIDERATIONS

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulator actions for categorical exclusion for performing an environmental assessment. A proposed change for an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; (3) result in an increase in individual or cumulative occupational radiation exposure. We have reviewed this request and determined that the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance with the amendment. The basis for this determination follows.

Proposed Change

This request proposes to change the technical specifications to be consistent with NUREG-1431; Standard Technical Specifications, Westinghouse Plants Revision 1, 04/07/95 within limitations imposed by plant specific design and licensing basis.

Basis

The proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons.

1. As demonstrated in the No Significant Hazards Evaluation, the proposed changes do not involve a significant hazards consideration.

2. These proposed changes are being made to establish consistency with the improved Standard Technical Specifications (ISTS) - Westinghouse Plants, NUREG 1431, Rev. 1, including approved generic changes, and do not involve physical changes to the facility, nor do they affect actual plant effluents.

3. These proposed changes are being made to establish consistency with the improved Standard Technical Specifications (ISTS) - Westinghouse Plants, NUREG 1431, Rev. 1, including approved generic changes and do not involve physical changes to the facility, and they do not significantly affect individual or cumulative occupational radiation exposures.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.0 - LCO AND SR APPLICABILITY

PART 4

MARKUP OF NUREG 1431, REVISION 1, "STANDARD TECHNICAL SPECIFICATIONS - WESTINGHOUSE PLANTS" (ISTS)

1

CTS

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

[A2] LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and 3.0.7 TSTF-6.

[A3] LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

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

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

a. MODE 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

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

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

[M2] LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This

(continued)

CTS

3.0 LCO APPLICABILITY

1

LCO 3.0.4 (continued)

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

Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.

2

LCO 3.0.4 ~~is~~ only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1. 2. 3. and 4.

~~Reviewers's Note: LCO 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, LCO 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1. 2. 3. and 4. The MODE change restrictions in LCO 3.0.4 were previously applicable in all MODES. Before this version of LCO 3.0.4 can be implemented on a plant-specific basis, the licensee must review the existing technical specifications to determine where specific restrictions on MODE changes or Required Actions should be included in individual LCOs to justify this change; such an evaluation should be summarized in a matrix of all existing LCOs to facilitate NRC staff review of a conversion to the STS.~~

[A5]

LCO 3.0.5

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

(continued)

CTS

3.0 LCO APPLICABILITY (continued)

[A6] LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.5.15, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

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

[A7] LCO 3.0.7 Test Exception LCOs 3.1.8, 3.1.10, 3.1.11, and 3.4.19 allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

(continued)

CTS

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

[4.0]

SR 3.0.1

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

[4.0]

SR 3.0.2

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

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

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

Exceptions to this Specification are stated in the individual Specifications.

[4.0]

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

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

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be

(continued)

CTS

3.0 SR APPLICABILITY

SR 3.0.3 (continued) declared not met, and the applicable Condition(s) must be entered.

[M2]

SR 3.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3 and 4.

~~Reviewer's Note: SR 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, SR 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4. The MODE change restrictions in SR 3.0.4 were previously applicable in all MODES. Before this version of SR 3.0.4 can be implemented on a plant-specific basis, the licensee must review the existing technical specifications to determine where specific restrictions on MODE changes or Required Actions should be included in individual LCOs to justify this change; such an evaluation should be summarized in a matrix of all existing LCOs to facilitate NRC staff review of a conversion to the STS.~~

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.0 - LCO AND SR APPLICABILITY

PART 5

JUSTIFICATION FOR DIFFERENCES (JFDs) TO ISTS

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes which involve the insertion of plant specific terms or parameters are used to preserve consistency with the CTS and licensing basis.

2 Specification presentation is modified for clarity, or to correct typographical or grammatical errors.

3 ISTS Specification 3.4.19 is not adopted in the ITS. Natural circulation tests are neither required, nor performed, as part of low power physics testing. Operation in MODES 1 and 2 with less than three RCS Loops in operation is not permitted and is no longer analyzed for each fuel cycle.

4 ISTS LCO 3.1.9 and LCO 3.1.11 are deleted. ISTS LCO 3.1.10 is renumbered to 3.1.9. The physic test that LCO 3.1.9 required are RCCA Pseudo Ejection Test, RCCA Pseudo Drop and Misalignment Test, and Xenon Stability Measurements. These physics tests were only performed in some initial plant startup testing. These tests can be deleted since these physics tests are not performed during post-refueling startup testing. The physics test that LCO 3.1.11 requires was the Rod Worth Measurement in the N minus 1 condition. The use of other rod worth measurement techniques will maintain the shutdown margin during the entire measurement process and still provide the necessary physics data verification. Since the N minus 1 measurement technique is no longer used, the SDM test exception can be deleted.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.0 - LCO AND SR APPLICABILITY

PART 6

MARKUP OF ISTS BASES

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.6 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

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

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

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

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering

(continued)

BASES

LCO 3.0.2 (continued)

ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

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

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

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

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

(continued)

BASES (continued)

LCO 3.0.3

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

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

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

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

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

(continued)

BASES

LCO 3.0.3 (continued)

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

a. The LCO is now met.

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

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

The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.15, "Fuel Storage Pool Water Level." LCO 3.7.15 has an Applicability of "During movement of irradiated fuel

(continued)

BASES

LCO 3.0.3 (continued)

assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.13 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.12 of "Suspend movement of irradiated fuel assemblies in the fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability

(continued)

BASES

1

LCO 3.0.4 (continued)

that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. [In some cases (e.g., ...) these ACTIONS provide a Note that states "While this LCO is not met, entry into a MODE or other specified condition in the Applicability is not permitted, unless required to comply with ACTIONS." This Note is a requirement explicitly precluding entry into a MODE or other specified condition of the Applicability.]

4

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to

(continued)

BASES

LCO 3.0.5 (continued) - provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

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

b. The OPERABILITY of other equipment.

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

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

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

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

(continued)

BASES

1

LCO 3.0.6 (continued)

supported system's Conditions and Required Actions or may specify other Required Actions.

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

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

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support

(continued)

BASES

LCO 3.0.6 (continued)

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

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs 3.1.8, 3.1.9, 3.1.18, 3.1.11, and 3.4.19 allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

1

CTS

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

[4.0]

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

(continued)

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . " interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is a Surveillance with a Frequency of "in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions." The requirements of regulations take precedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations.

(continued)

BASES

SR 3.0.2 (continued)

Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most

(continued)

BASES

SR 3.0.3 (continued)

probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

(continued)

BASES

SR 3.0.4 (continued)

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

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event.

(continued)

BASES

SR 3.0.4 (continued)

condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, Mode 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.0 - LCO AND SR APPLICABILITY

PART 7

JUSTIFICATION FOR DIFFERENCES (JFDs) TO ISTS BASES

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes which involve the insertion of plant specific terms or parameters are used to preserve consistency with the CTS and licensing basis.

2 Bases presentation is modified for clarity, or to correct typographical or grammatical errors.

3 In the Bases for ISTS LCO 3.0.3, the term "Specification" is changed to "LCO" to clarify that the requirements of 3.0.3 are limiting conditions for operation, and to be consistent with usage of terms elsewhere in the ISTS. This change is administrative and therefore has no adverse impact on safety.

4 The bracketed information in the Bases for LCO 3.0.4 are deleted, since the material discussed is not used in the ITS.

5 ISTS Specification 3.4.19 is not adopted in the ITS. Natural circulation tests are neither required, nor performed, as part of low power physics testing. Operation in MODES 1 and 2 with less than three RCS Loops in operation is not permitted and is no longer analyzed for each fuel cycle.

6 ISTS LCO 3.1.9 and LCO 3.1.11 are deleted. ISTS LCO 3.1.10 is renumbered to 3.1.9. The physic test that LCO 3.1.9 required are RCCA Pseudo Ejection Test, RCCA Pseudo Drop and Misalignment Test, and Xenon Stability Measurements. These physics tests were only performed in some initial plant startup testing. These tests can be deleted since these physics tests are not performed during post-refueling startup testing. The physics test that LCO 3.1.11 requires was the Rod Worth Measurement in the N minus 1 condition. The use of other rod worth measurement techniques will maintain the shutdown margin during the entire measurement process and still provide the necessary physics data verification. Since the N minus 1 measurement technique is no longer used, the SDM test exception can be deleted.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.0 - LCO AND SR APPLICABILITY

PART 8

PROPOSED HBRSEP, UNIT NO. 2 ITS

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

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

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

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

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

a. MODE 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

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

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

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

Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

LCO 3.0.5

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

LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.5.15, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.6 (continued)	applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.
LCO 3.0.7	Test Exception LCO 3.1.8 allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.0 - LCO AND SR APPLICABILITY

PART 9

PROPOSED BASES TO HBRSEP, UNIT NO. 2 ITS

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.6 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified. There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering

(continued)

BASES

LCO 3.0.2 (continued)

ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

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

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Condition no longer exists. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

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

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

(continued)

BASES

LCO 3.0.2 (continued)

becomes applicable, and the ACTIONS Condition(s) are entered.

LCO 3.0.3

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

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

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

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

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on

(continued)

BASES

LCO 3.0.3 (continued)

components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

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

a. The LCO is now met.

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

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

The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

(continued)

BASES

LCO 3.0.3 (continued)

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.12, "Fuel Storage Pool Water Level." LCO 3.7.12 has an Applicability of "During movement of irradiated fuel assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.12 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.12 of "Suspend movement of irradiated fuel assemblies in the fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good

(continued)

BASES

LCO 3.0.4 (continued)

practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to

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BASES

LCO 3.0.5 (continued)

provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

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

b. The OPERABILITY of other equipment.

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

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

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

LCO 3.0.6

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

(continued)

BASES

LCO 3.0.6 (continued)

specify other Required Actions.

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

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

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant

(continued)

BASES

LCO 3.0.6 (continued)

safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCO 3.1.8 allows specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

(continued)

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is a Surveillance with a Frequency of "in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions." The requirements of regulations take precedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations.

(continued)

BASES

SR 3.0.2 (continued)

Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most

(continued)

BASES

SR 3.0.3 (continued)

probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a delay period for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

(continued)

BASES

SR 3.0.4 (continued)

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

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a mode change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefor, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event,

(continued)

BASES

SR 3.0.4 (continued)

condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.0 - LCO AND SR APPLICABILITY

PART 10

ISTS GENERIC CHANGES

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
CONVERSION PACKAGE SECTION 3.1

ITS CONVERSION PACKAGE

CHAPTER 3.1 - REACTIVITY CONTROL SYSTEMS

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.1 - REACTIVITY CONTROL SYSTEMS

PART 1

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

ITS

A1

3.10.6.2 No more than one inoperable control rod shall be permitted during power operation.

See LCO 3.1.4

3.10.6.3 If a full length control rod cannot be moved by its mechanism, boron concentration shall be changed to compensate for the withdrawn worth of the inoperable rod such that a shutdown margin equal to or greater than shown on Figure 3.10-2 results.

3.10.7 Power Ramp Rate Limits

3.10.7.1 During the return to power following a shutdown where fuel assemblies have been handled (e.g., refueling, inspection), the rate of reactor power increase shall be limited to 3 percent of rated power in an hour between 20 percent and 100 percent of rated power. This ramp rate requirement applies during the initial startup and may apply during subsequent power increases, depending on the maximum power level achieved and length of operation at that power level. Specifically, this requirement can be moved for reactor power levels below a power level P (20 percent < P ≤ 100 percent), provided that the plant has operated at or above power level P for at least 72 cumulative hours out of any seven-day operating period following the shutdown.

R1

3.10.7.2 The rate of reactor power increases above the highest power level sustained for at least 72 cumulative hours during the preceding 30 cumulative days of reactor power operation shall be limited to 3 percent of rated power in an hour. Alternatively, reactor power increase can be accomplished by a single step increase less than or equal to 10 percent of rated power followed by a maximum ramp rate of 3 percent of rated power in an hour beginning three hours after the step increase.

3.10.8 Required Shutdown Margins

MODE 2 with Keff < 1.0
MODES 3, 4

A2

[LCO 3.1.1]

[Applicability]

3.10.8.1 When the reactor is in the ~~hot shutdown~~ condition, the shutdown margin shall be ~~at least that shown in Figure 3.10-2.~~

within the limits specified in the COLR

LA1

SPECIFICATION 3.1.1

ITS

A1

A2

MODE 5

[LCO 3.1.1]
[Applicability]

3.10.8.2 When the reactor is in the cold shutdown condition, the shutdown margin shall be at least 1 percent Δk/k within the limits specified in the COLR

LA1

See LCO 3.9.1

3.10.8.3 When the reactor is in the refueling operation mode, the shutdown margin shall be at least 6 percent Δk/k.

~~Basis~~

The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of reactor power is by the control groups. A reactor trip occurring during power operation will put the reactor into the hot shutdown condition.

The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time assuming the highest worth control rod remains fully withdrawn with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of hypothetical rod ejection and provide for acceptable nuclear peaking factors. The control rod insertion limits are specified in the CORE OPERATING LIMITS REPORT (COLR) and are appropriately chosen to meet the shutdown requirements shown in Figure 3.10-2. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in analysis of the hypothetical steam break accident. Early in core life, less shutdown margin is required, and Figure 3.10-2 shows the shutdown margin required at end of life with respect to an uncontrolled cooldown. All other accident analyses are based on 1 percent reactivity

A7

Add RA A.1 — L1

Add SR 3.1.1.1 — M1

Specication 3.1.1

LA1

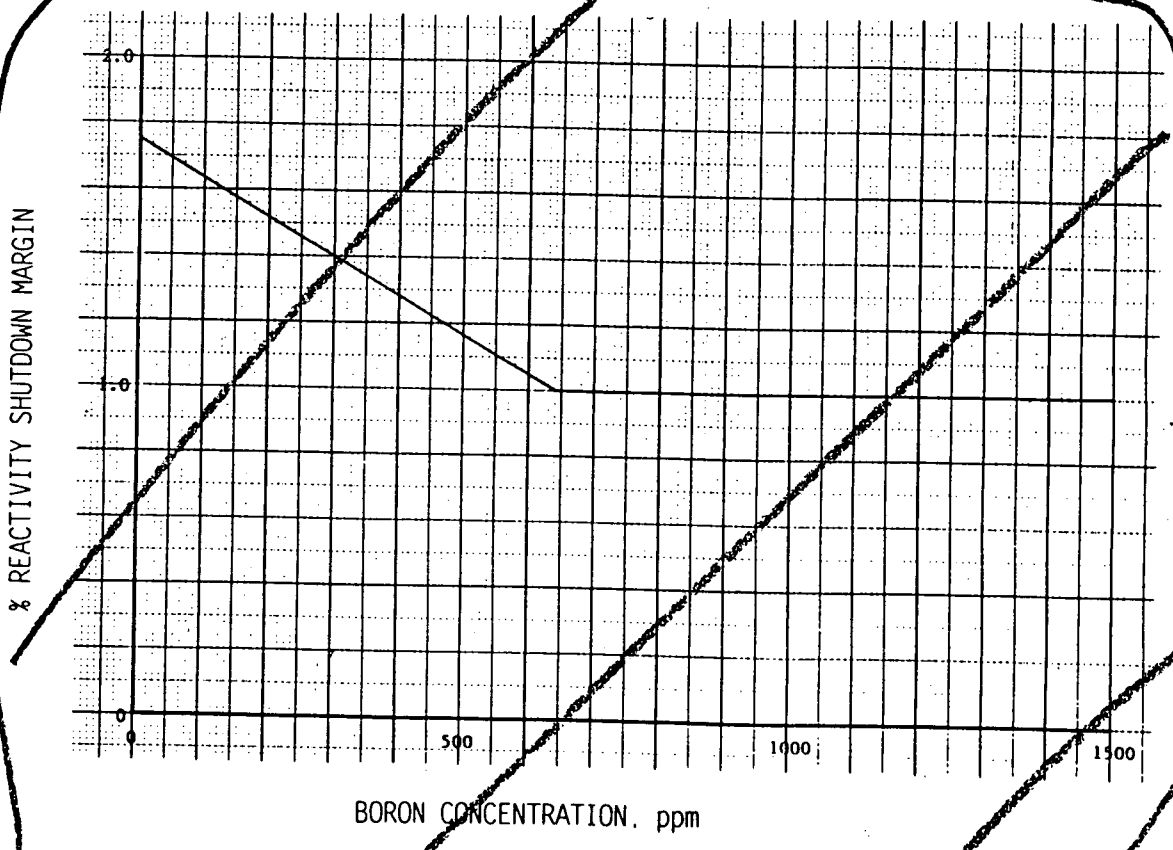

Figure 3.10-2 SHUTDOWN MARGIN VS BORON CONCENTRATION

Specification 3.1.2

ITS

A1

4.9 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of reactivity anomalies within the reactor.

Specification

M2

[SR 3.1.2.1] Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, submit a Special Report to the Commission within 30 days.

A3

Basis

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should

A7

Add LCO 3.1.2 + Applicability
RAs A.1, A.2 + B.1

M3

Specification 3.1.3

ITS

A1

See 3.1.8

3.1.3 Minimum Conditions for Criticality

MODE 1 + MODE 2 w/ Keff ≥ 1.0

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature at which the moderator temperature coefficient is outside the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be less than or equal to:

[LCO 3.1.3]
[Applicability]

a) +5.0 pcm/°F at less than 50% of rated power, or

b) 0 pcm/°F at 50% of rated power and above.

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1.

See 3.4.2

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is outside the limits specified in the COLR, the reactor shall be made subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

[RA B.1]
[RA A.1]

L2

3.1.3.4 The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

See 3.4.9

Basis

During the early part of fuel cycle, the moderator temperature coefficient may be slightly positive at low power levels. The moderator temperature coefficient at low temperatures or powers will be most positive at the beginning of the fuel cycle, when the boron concentration in the coolant is the greatest. At all times, the moderator temperature coefficient is calculated to be negative in the high power operating range, and after a very brief period of power operation, the coefficient will be negative in all circumstances due to the reduced boron concentration as Xenon and fission products build into the core. The requirement that the reactor is not to be made critical when the moderator temperature coefficient outside the limits specified in the COLR has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase in moderator temperature or decrease in coolant pressure. This requirement is

A7

Add Applicability for MTC lower limit

M4

Add RA C.1

M5

Add SR 3.1.3.1
SR 3.1.3.2

M6

Specification 3.1.4 (A1)

ITS

See - 3.1.8

All shutdown and

shall be OPERABLE

will all individual indicated rod positions (S)

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows:

[LCO 3.1.4]

- at positions ≥ 200 steps and is > 15 inches out of alignment with its bank position, or

- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

(A4)

(M7)

then within two hours, perform the following:

ONE

[RAB.1] a. Correct the situation, or

within 72 hours perform SR 3.2.1.1 + 3.2.2.1

[RAB2.4] [RAB.2.5] b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1 or

and

(M8)

Reduce Thermal

[RAB.2.2] c. Limit power to ≤ 70 percent of rated power

3.10.1.6 Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted.

[NOTE TO LCO 3.1.5 AND LCO 3.1.6]

See 3.1.5 3.1.6

See 3.1.8

3.10.2 Power Distribution Limits

3.10.2.1 At all times except during low power physics tests, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the basis, must meet the following limits:

$$F_Q(Z) \leq (F_Q^{RTP}/P) \times K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) < (F_Q^{RTP}/0.5) \times K(Z) \text{ for } P \leq 0.5$$

$$F_{\Delta H} < F_{\Delta H}^{RTP} \ (1 + PF_{\Delta H} \ (1-P) \ )$$

See 3.2.1 + 3.2.2

Add MODES 1 + 2 (A5)

Add RA B.2.1.1, B.2.1.2
RA B.2.3, B2.6
RA C.1, D.1.1, D.1.2
RA D.2

(M9)

ITS

A1

3.10.4 Rod Drop Time

[SR 3.1.4.3] 3.10.4.1 The drop time of each control rod shall be not greater than 1.8 seconds at full flow and operating temperature from the beginning of rod motion to dashpot entry.

from the fully withdrawn position is (A8) verify the rod ≤ >540°F with all reactor coolant pumps operating

3.10.5 Reactor Trip Breakers

3.10.5.1 The reactor shall not be made critical unless the following conditions are met:

a. Two reactor trip breakers are operable.

b. Reactor trip bypass breakers are racked out or removed.

c. Two trains of automatic trip logic are operable.

Decay of stationary gripper coil Voltage (M24)

3.10.5.2 During power operation, the requirements of 3.10.5.1 may be modified to allow the following components to be inoperable. If the system is not restored to meet the requirements of 3.10.5.1, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures within the next 8 hours.

a. One reactor trip breaker may be inoperable for up to 12 hours.

b. One train of automatic trip logic may be inoperable for up to 12 hours.

c. One reactor trip bypass breaker may be racked in and closed for up to 12 hours.

see 3.3.1

3.10.5.3 With one of the diverse trip features inoperable (shunt trip attachment/undervoltage trip attachment) on one of the reactor trip breakers, power operation may continue for up to 48 hours. If the

Add Condition A, associated actions, and completion times for control rod drop time not within limits (M.26)

SPECIFICATION
3.1.4

ITS

diverse trip feature is not restored to operable status within this time, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures within the next eight hours.

see 3.3.1

3.10.6 Inoperable Control Rods

[LCO 3.1.4]

3.10.6.1 A control rod shall be deemed inoperable if (a) the rod is misaligned by more than 15 inches with its bank, (b) ~~if the rod cannot be moved by its drive mechanism~~, or (c) if its ~~rod drop time is not met~~.

A6

[SR 3.1.4.2]

[SR 3.1.4.3]

ITS

SPECIFICATION 3.1.4

A1

M10

3.10.6.2 No more than one inoperable control rod shall be permitted during power operation.

[RA A.1.1]
[RA A.1.2]

3.10.6.3 If a full length control rod cannot be moved by ~~its mechanism~~ (one or more) (s) inoperable, the boron concentration shall be changed to compensate for the withdrawn worth of the inoperable rod such that a shutdown margin equal to or greater than shown on Figure 3.10-2 results.

to restore SDM within limit in 1 hour

or verify SDM is within limit specified in the COLR in 1 hour

M25

LA 1

3.10.7 Power Ramp Rate Limits

3.10.7.1 During the return to power following a shutdown where fuel assemblies have been handled (e.g., refueling, inspection), the rate of reactor power increase shall be limited to 3 percent of rated power in an hour between 20 percent and 100 percent of rated power. This ramp rate requirement applies during the initial startup and may apply during subsequent power increases, depending on the maximum power level achieved and length of operation at that power level. Specifically, this requirement can be moved for reactor power levels below a power level P (20 percent < P ≤ 100 percent), provided that the plant has operated at or above power level P for at least 72 cumulative hours out of any seven-day operating period following the shutdown.

See 3.1.1

3.10.7.2 The rate of reactor power increases above the highest power level sustained for at least 72 cumulative hours during the preceding 30 cumulative days of reactor power operation shall be limited to 3 percent of rated power in an hour. Alternatively, reactor power increase can be accomplished by a single step increase less than or equal to 10 percent of rated power followed by a maximum ramp rate of 3 percent of rated power in an hour beginning three hours after the step increase.

3.10.8 Required Shutdown Margins

3.10.8.1 When the reactor is in the hot shutdown condition, the shutdown margin shall be at least that shown in Figure 3.10-2.

See 3.1.1

M25

Add RA A.2

ITS

Specification 3.1.4

A1

TABLE 4.1-3

FREQUENCIES FOR EQUIPMENT TESTS

L3

		Check	Frequency	Maximum Time Between Tests	
[SR 3.1.4.3] 1.	Control Rods	Rod drop times of all full length rods	Each refueling shutdown	NA	
[SR 3.1.4.2] 2.	Control Rod Not Fully Inserted	≥ 10 Steps ~~Partial~~ movement of all full length rods	~~Every 2 weeks during reactor critical operations~~ 92 days	~~20 days~~	M11 L3
3.	Pressurizer Safety Valves	Set point	Each refueling shutdown	NA	see 3.4.10
4.	Main Steam Safety Valves	Verify each required MSSV lift setpoint per Table 4.1-4 in accordance with the Inservice Testing Program. Following testing, lift setting shall be within +/- 1%.	~~In accordance with~~ the Inservice Testing Program	NA	see 3.7.1
5.	Containment Isolation Trip	Functioning	Each refueling shutdown	NA	see 3.6.3 + 3.3.2
6.	Refueling System Interlocks	Functioning	Prior to each refueling shutdown	NA	see 3.9.1
7.	Service Water System	Functioning	Each refueling shutdown	NA	see 3.7.7
~~8.~~	~~DELETED~~				
9.	Primary System Leakage	Evaluate	Daily when reactor coolant system is above cold shutdown condition	NA	see 3.4.13
10.	Diesel Fuel Supply	Fuel Inventory	Weekly	10 days	see 3.8.3
~~11.~~	~~DELETED~~				
12.	Turbine Steam Stop, Control, Reheat Stop, and Interceptor Valves	Closure	Quarterly during power operation and prior to startup	115 days	see 3.7.1

And SR 3.1.4.1 — M12

LAR dated 1/29/96

~~Amendment No. 142, 155, 164~~

ITS

Specification 3.1.5

A1

3.10 REQUIRED SHUTDOWN MARGINS, CONTROL ROD, AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the required shutdown margins, operation of the control rods, and power distribution limits.

Objective

To ensure (1) core subcriticality after a reactor trip and during normal shutdown conditions, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

M13

Specification

3.10.1 Full Length Control Rod Insertion Limits

3.10.1.1 (Deleted by Change No. 21 issued 7/6/73)

3.10.1.2 When the reactor is critical, except for physics tests and full length control rod exercises, the shutdown control rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

[Applicability]

[LCO 3.1.5]

IN MODE 1 and MODE 2 with any Control Rod Not Fully Inserted

See 3.1.8

bank

3.10.1.3 When the reactor is critical, except for physics tests and full length control rod exercises, the control rods shall be limited in physical insertion as specified in the COLR. Control rod bank insertion beyond the limits specified in the COLR shall be corrected within the time criteria established by the axial power distribution methodology or within one (1) hour, whichever occurs sooner. If bank insertion is not restored to the specified limits (i.e., within one (1) hour or within the time criteria established by the axial power distribution methodology, whichever is sooner) the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures within six (6) hours.

3.10.1.4 At 50 percent of the cycle as defined by burnup, the limits shall be adjusted to the end-of-core values as specified in the COLR.

See 3.1.6

Add RAs A.1.1, A.1.2, A.2, B.1 — M14

Add SR 3.1.5.1 — M15

Amendment No. ~~141~~, 167

Specification 3.1.5

ITS

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows:

- at positions ≥ 200 steps and is > 15 inches out of alignment with its bank position, or
- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

See 3.1.4

then within two hours, perform the following:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power

See 3.1.8

[3.1.5] [Applicability Note]

3.10.1.6 Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted.

See 3.1.8

3.10.2 Power Distribution Limits

3.10.2.1 At all times except during low power physics tests, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the basis, must meet the following limits:

$F_Q(Z) \leq (F_Q^{RTP}/P) \times K(Z)$ for $P > 0.5$

$F_Q(Z) < (F_Q^{RTP}/0.5) \times K(Z)$ for $P \leq 0.5$

$F_{\Delta H} < F_{\Delta H}^{RTP} (1 + PF_{\Delta H} (1-P))$

See 3.2.1 3.2.2

ITS

Specification 3.1.6

3.10 REQUIRED SHUTDOWN MARGINS, CONTROL ROD, AND POWER DISTRIBUTION LIMITS

A1

~~Applicability~~

~~Applies to the required shutdown margins, operation of the control rods, and power distribution limits.~~

~~Objective~~

~~To ensure (1) core subcriticality after a reactor trip and during normal shutdown conditions, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.~~

Specification

3.10.1 Full Length Control Rod Insertion Limits

~~3.10.1.1 (Deleted by Change No. 21 issued 7/6/73)~~

3.10.1.2 When the reactor is critical, except for physics tests and full length control rod exercises, the shutdown control rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

See 3.1.5 + 3.1.8

[Applicability]

MODE 1 / MODE 2 with $K_{eff} \geq 1.0$

BANKS

See 3.1.8

3.10.1.3 When the reactor is critical, except for physics tests and full length control rod exercises, the control rods shall be limited in physical insertion as specified in the COLR. Control rod bank insertion beyond the limits specified in the COLR shall be corrected within the time criteria established by the axial power distribution methodology or within one (1) hour, whichever occurs sooner. If bank insertion is not restored to the specified limits (i.e., within one (1) hour or within the time criteria established by the axial power distribution methodology, whichever is sooner) the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures within six (6) hours.

[LCO 3.1.6]

[RA A.2]

[RA C.1]

within

2

L4

LA2

3.10.1.4 At 50 percent of the cycle as defined by burnup, the limits shall be adjusted to the end-of-core values as specified in the COLR.

LA1

the insertion, sequence and overlap

M16

Add RAs A.1.1, A1.2, B.1.1, B.1.2 + B.2

M17

Add SRs 3.1.6.1
3.1.6.2
3.1.6.3

M18

Amendment No. ~~141~~, 167

Specification 3.1.6

ITS

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows:

- at positions ≥ 200 steps and is > 15 inches out of alignment with its bank position, or
- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

See 3.1.4

then within two hours, perform the following:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power

See 3.1.8

[3.1.6] [Applicability Note]

3.10.1.6 Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted.

See 3.1.8

3.10.2 Power Distribution Limits

3.10.2.1 At all times except during low power physics tests, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the basis, must meet the following limits:

$F_Q(Z) \leq (F_Q^{RTP}/P) \times K(Z)$ for $P > 0.5$

$F_Q(Z) < (F_Q^{RTP}/0.5) \times K(Z)$ for $P \leq 0.5$

$F_{\Delta H} < F_{\Delta H}^{RTP} (1 + PF_{\Delta H} (1-P))$

See 3.2.1 3.2.2

Specification 3.1.7

3.10 REQUIRED SHUTDOWN MARGINS, CONTROL ROD, AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the required shutdown margins, operation of the control rods, and power distribution limits.

Objective

To ensure (1) core subcriticality after a reactor trip and during normal shutdown conditions, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

Specification

3.10.1 Full Length Control Rod Insertion Limits — See 3.1.5 3.1.6 3.1.8

3.10.1.1 (Deleted by Change No. 21 issued 7/6/73)

3.10.1.2 When the reactor is critical, except for physics tests and full length control rod exercises, the shutdown control rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

3.10.1.3 When the reactor is critical, except for physics tests and full length control rod exercises, the control rods shall be limited in physical insertion as specified in the COLR. Control rod bank insertion beyond the limits specified in the COLR shall be corrected within the time criteria established by the axial power distribution methodology or within one (1) hour, whichever occurs sooner. If bank insertion is not restored to the specified limits (i.e., within one (1) hour or within the time criteria established by the axial power distribution methodology, whichever is sooner) the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures within six (6) hours.

3.10.1.4 At 50 percent of the cycle as defined by burnup, the limits shall be adjusted to the end-of-core values as specified in the COLR.

Add Specification 3.1.7 — M19

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature, at which the moderator temperature coefficient is outside the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limits shall be less than or equal to:

a) +5.0 pcm/°F at less than 50% of rated power, or

b) 0 pcm/°F at 50% of rated power and above.

See 3.1.3

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1.

See 3.4.2

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is outside the limits specified in the COLR, the reactor shall be made subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

See 3.1.3

3.1.3.4 The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

See 3.4.9

Basis

During the early part of fuel cycle, the moderator temperature coefficient may be slightly positive at low power levels. The moderator temperature coefficient at low temperatures or powers will be most positive at the beginning of the fuel cycle, when the boron concentration in the coolant is the greatest. At all times, the moderator temperature coefficient is calculated to be negative in the high power operating range, and after a very brief period of power operation, the coefficient will be negative in all circumstances due to the reduced boron concentration as Xenon and fission products build into the core. The requirement that the reactor is not to be made critical when the moderator temperature coefficient outside the limits specified in the COLR has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase in moderator temperature or decrease in coolant pressure. This requirement is

A7

M22

Add Applicability During PHYSICS TESTS

Specification 3.1.8

A1

3.10 REQUIRED SHUTDOWN MARGINS, CONTROL ROD, AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the required shutdown margins, operation of the control rods, and power distribution limits.

Objective

To ensure (1) core subcriticality after a reactor trip and during normal shutdown conditions, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

Specification

3.10.1 Full Length Control Rod Insertion Limits

3.10.1.1 (Deleted by Change No. 21 issued 7/6/73)

[LCO 3.1.8]

3.10.1.2 When the reactor is critical, except for physics tests ~~and~~ full length control rod exercises, the shutdown control rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

See 3.1.5 3.1.6

[LCO 3.1.8]

3.10.1.3 When the reactor is critical, except for physics tests ~~and~~ full length control rod exercises, the control rods shall be limited in physical insertion as specified in the COLR. Control rod bank insertion beyond the limits specified in the COLR shall be corrected within the time criteria established by the axial power distribution methodology or within one (1) hour, whichever occurs sooner. If bank insertion is not restored to the specified limits (i.e., within one (1) hour or within the time criteria established by the axial power distribution methodology, whichever is sooner) the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures within six (6) hours.

3.10.1.4 At 50 percent of the cycle as defined by burnup, the limits shall be adjusted to the end-of-core values as specified in the COLR.

Add RAs A.1, A.2, B.1, C.1, D.1 — M20

Add SR 3.1.8.1
SR 3.1.8.2
SR 3.1.8.3 — M21

Add LCO 3.1.8 requirements a, b, and c — M22

ITS

[LCO 3.1.8]

Specification 3.1.8

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows:

- at positions ≥ 200 steps and is > 15 inches out of alignment with its bank position, or
- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

then within two hours, perform the following:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power

See 3.1.4

[LCO 3.1.8]

3.10.1.6 Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted.

See 3.1.5 3.1.6

M23

3.10.2 Power Distribution Limits

3.10.2.1 At all times except during low power physics tests, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the basis, must meet the following limits:

$$F_Q(Z) \leq (F_Q^{RTP}/P) \times K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) < (F_Q^{RTP}/0.5) \times K(Z) \text{ for } P \leq 0.5$$

$$F_{\Delta H} < F_{\Delta H}^{RTP} (1 + PF_{\Delta H} (1-P))$$

See 3.2.1 3.2.2

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.1 - REACTIVITY CONTROL SYSTEMS

PART 2

DISCUSSION OF CHANGES (DOCS) FOR CTS MARKUP

ADMINISTRATIVE CHANGES

A1 In the conversion of the H.B. Robinson Steam Electric Plant (HBRSEP), Unit No.2 Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in the - Standard Technical Specifications, Westinghouse Plants NUREG-1431, Rev 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS 3.10.8.1 requires that the shutdown margin (SDM) be within the limits specified in figure 3.10-2 when the reactor is in hot shutdown. CTS 3.10.8.2 requires that the SDM be at least 1 percent $\Delta k/k$ when the reactor is in cold shutdown. ITS 3.1.1 specifies that the SDM be maintained within the limits provided in the Core Operating Limits Report (COLR) with a specified applicability of MODE 2 with k_{eff} <1.0 and MODES 3, 4 and 5. The CTS definition of hot shutdown encompasses ITS MODE 2 with k_{eff} <1.0 and ITS MODES 3 and 4. The CTS definition of cold shutdown is consistent with ITS MODE 5. Therefore, this is an administrative change resulting from combining CTS Specifications 3.10.8.1 and 3.10.8.2.

A3 CTS 4.9 requires submittal of a Special Report within 30 days if the difference between observed and predicted steady-state boron concentration reaches the equivalent of 1 percent $\Delta k/k$. This requirement is not retained in the ITS. A 1 percent $\Delta k/k$ reactivity anomaly is reasonable equivalent to change of 100 ppm in boron concentration. A 100 ppm boron uncertainty is included in applicable HBRSEP safety analysis. Therefore, elimination of this requirement is an administrative change since reporting of the anomaly is encompassed in the requirements of 10 CFR 50.72 and 50.73. This change is consistent with NUREG-1431.

A4 In addition to rewording and other editorial changes necessary to the conversion (DOC A1 above) CTS 3.10.5 is reworded in the ITS to clarify current licensing basis requirements. The wording "for bank demand positions" is used to replace the less precise terminology in the introductory phrase for each of the two cases beginning with "at positions . . ." This substitution is necessary to remove ambiguity regarding whether these statements are referring to bank demand or actual rod positions. Additionally, the terminology ". . . with its bank demand position . . ." is used to replace the less precise term " . . . bank position . . ." for the condition of ≥ 200 steps. This substitution is made to clarify that, for this condition, the applicable reference position is bank demand position. For the condition < 200

steps, the terminology ". . . the average of the individual rod positions . . ." is substituted for ". . . the average if its bank position . . ." to clarify that, for this condition, the applicable reference position is the actual average rod position. These are administrative changes made to eliminate confusion and ambiguity regarding application of these requirements.

A5 CTS 3.10.1.5 and 3.10.6 do not include explicit operating condition applicability statements. ITS adds specific applicability of MODES 1 and 2. MODES 1 and 2 are reasonable interpretations of the implicit applicability of these specifications. This change is consistent with NUREG-1431.

A6 CTS 3.10.6.1 provides attributes of inoperable control rods and conversely of OPERABLE control rods. Verification of these attributes are included as SRs in ITS. Since the SRs are included in ITS, there is no need to include the a separate definition of inoperable rods. Therefore, this is an administrative change and is consistent with NUREG-1431.

A7 The CTS Bases are not retained in the ITS, but are replaced in their entirety. The ITS includes significantly expanded and improved Bases. The Bases do not define or impose any specific requirements but serve to explain, clarify and document the reasons (i.e., Bases) for the associated specification. The Bases are not part of the Technical Specifications required by 10 CFR 50.36. Therefore, this is an administrative change.

A8 CTS 3.10.4.1 establishes the parameters for rod drop time testing but does not specifically address the rod position just prior to testing. ITS SR 3.1.4.3 includes the requirement "from the fully withdrawn position" which is implied by the CTS but not specifically stated. This change is a clarification and provides additional information but does not change any technical requirements and is considered an administrative change and is consistent with NUREG-1431.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 No CTS surveillance requirement (SR) comparable to ITS SR 3.1.1.1 exists. This SR is considered a reasonable verification of the associated requirement. Therefore, the addition of this SR is a more restrictive change and is consistent with NUREG-1431.

M2 CTS 4.9 requires, after normalization, periodic comparison of actual boron concentration to predicted boron values. CTS 4.9 neither specifies when the normalization is required nor the periodicity of the surveillance. ITS SR 3.1.2.1 requires this comparison to be performed prior to entering MODE 1 after initial criticality and every 31 EFPD after 60 EFPD after each fuel loading. The ITS retains the provision for normalization but specifies the normalization is required to be completed prior to exceeding 60 EFPD. Therefore, the ITS SR 3.1.2.1 is more restrictive and is consistent with NUREG-1431.

M3 A CTS limiting condition for operation (LCO) comparable to ITS LCO 3.1.2 does not exist. ITS LCO 3.1.2 provides for limitations on anomalous reactivity conditions and is applicable in MODES 1 and 2. A CTS action comparable to ITS 3.1.2 Required Actions (RAs) A.1 and A.2 does not exist. ITS 3.1.2 RAs A.1 and A.2 delineate actions to be completed within 72 hours, including re-evaluation of core design and safety analysis, confirmation of acceptability for continued operation and establishment of appropriate operating restrictions and SRs. RA B.1 mandates placing the unit in MODE 3 if RAs A.1 and A.2 and the associated completion times are not met. The inclusion of this LCO including associated RAs and SRs is considered reasonable to ensure operation within the bounds of the applicable safety analysis. Therefore, these are additional restrictions on plant operation and are consistent with NUREG-1431.

M4 CTS 3.1.3.1 is a Minimum Conditions for Criticality which is equivalent to ITS MODE 1 and MODE 2 with $K_{eff} \geq 1.0$. For the upper MTC limit, the applicability of ITS 3.1.3 is MODE 1 and MODE 2 with $K_{eff} \geq 1.0$, which is equivalent to CTS. For the lower MTC limit, the applicability of ITS 3.1.3 is MODES 1, 2 and 3. This change adds the applicability for MODE 2 with $K_{eff} < 1.0$ and MODE 3 for the lower MTC limit. The inclusion of this applicability is considered reasonable to ensure operation within the bounds of the applicable safety analysis. Therefore, these are additional restrictions on plant operation and is consistent with NUREG-1431.

M5 Not Used.

M6 CTS surveillance requirements comparable to ITS SR 3.1.3.1 and SR 3.1.3.2 do not exist. These SRs are considered a reasonable verification of the associated requirements. Therefore, these are additional restrictions on plant operation and are consistent with NUREG-1431.

M7 CTS 3.10.1.5 permits 2 hours to restore rod to within applicable limits. ITS only permits 1 hour. Therefore, this is a more restrictive change and is consistent with NUREG-1431.

M8 With out-of-alignment rods, CTS 3.10.1.5 requires that within 2 hours, either measurement and assessment of hot channel factors or a reduction of power to ≤ 70 percent rated thermal power be completed. ITS requires both reduction of power to less than 70 percent rated thermal power (within 2 hours) and completion of the surveillance of hot channel factors (within 72 hours). The inclusion of both of these actions is considered reasonable to ensure operation within the bounds of the applicable safety analysis. Therefore, this is an additional restriction on plant operation and is consistent with NUREG-1431.

M9 CTS does not include actions comparable to ITS 3.1.4 RAs B.2.1.1, B.2.1.2, B.2.3, B.2.6, C.1, D.1.1, D.1.2 and D.2. The inclusion of these RAs is considered reasonable to ensure operation within the bounds of the applicable safety analysis. Therefore, these are additional restrictions on plant operation and are consistent with NUREG-1431.

M10 CTS 3.10.6.2 permits operation with one inoperable rod. This specification is not retained in ITS. The elimination of this allowance ensures operation within the bounds of the applicable safety analysis. Therefore, this is an additional restriction on plant operation and is consistent with NUREG-1431.

M11 CTS Table 4.1-3, Item 2 specifies partial control rod movement. ITS SR 3.1.4.2 mandates ≥ 10 steps in either direction. Since minimum requirements for rod movement are being added, this change is more restrictive. This is an additional restriction on plant operation and is consistent with NUREG-1431.

M12 A CTS surveillance requirement comparable to ITS SR 3.1.4.1 does not exist. This SR is considered a reasonable verification of the associated requirement. Therefore, this is an additional restriction on plant operation and is consistent with NUREG-1431.

M13 CTS 3.10.1.2 is applicable when the reactor is critical. ITS 3.1.5 is applicable in MODE 1 and MODE 2 with any control bank not fully inserted. The inclusion of this additional applicability ensures operation within the bounds of the applicable safety analysis. This is

an additional restriction on plant operation and is consistent with NUREG-1431.

M14 CTS required actions comparable to ITS 3.1.5 RAs A.1.1, A.1.2, A.2 and B.1 do not exist. Lacking specified actions, failure to satisfy CTS 3.10.1.2 requires compliance with CTS 3.0. In this case CTS 3.0 requires hot shutdown in 8 hours. ITS 3.1.5 RAs A.1.1, A.1.2, A.2 and B.1 mandate actions to be in Hot Shutdown in 8 hours, the same as CTS 3.0. However, the additional actions in A.1.1 and A.1.2 are more restrictive on plant operation and are consistent with NUREG-1431.

M15 A CTS surveillance comparable to ITS SR 3.1.5.1 does not exist. This SR is considered a reasonable verification of the associated requirement. Therefore, the addition of ITS SR 3.1.5.1 is an additional restriction on plant operation and is consistent with NUREG-1431.

M16 CTS 3.10.1.3 does not impose explicit restrictions on sequence and overlap. These restrictions are explicitly incorporated in the ITS. The inclusion of these restrictions is considered reasonable to ensure operation within the bounds of the applicable safety analysis. These are additional restrictions on plant operation and are consistent with NUREG-1431.

M17 CTS required actions comparable to ITS 3.1.6 RAs A.1.1, A.1.2, B.1.1, B.1.2, and B.2 do not exist. The inclusion of these RAs is considered reasonable to ensure operation within the bounds of the applicable safety analysis. These are additional restrictions on plant operation and are consistent with NUREG-1431.

M18 CTS surveillance requirements comparable to ITS SRs 3.1.6.1, 3.1.6.2 and 3.1.6.3 do not exist. These SRs are considered a reasonable verification of the associated requirements. Therefore, the addition of these SRs is an additional restriction on plant operation and is consistent with NUREG-1431.

M19 A CTS comparable to ITS Specification 3.1.7 does not exist. The addition of this specification is considered reasonable to ensure operation within the bounds of the applicable safety analysis. Therefore, the addition of ITS specification 3.1.7 is an additional restriction on plant operation and is consistent with NUREG-1431.

M20 CTS required actions comparable to ITS 3.1.8 RAs A.1, A.2, B.1, C.1 and D.1 currently do not exist. Lacking specified actions, failure to satisfy CTS 3.1.3.1, 3.10.1.2, 3.10.1.3 or 3.10.1.5 requires compliance with CTS 3.0. CTS 3.0 requires hot shutdown in 8 hours and cold shutdown in 30 hours. ITS 3.1.8 RAs A.1, A.2, B.1, C.1 and D.1 mandate actions in shorter times (i.e., either immediately, 15 minutes or 1

hour). These are additional restrictions on plant operation and are consistent with NUREG-1431.

M21 CTS surveillance requirements comparable to ITS SRs 3.1.8.1, 3.1.8.2, and 3.1.8.3 do not exist. These SRs are considered a reasonable verification of the associated requirements. Therefore, the adoption of these SRs is an additional restriction on plant operation and is consistent with NUREG-1431.

M22 Physics tests exceptions included in CTS 3.1.3.1, 3.10.1.2, 3.10.1.3 and 3.10.1.5 do not specify any additional restriction when applying the exception. ITS 3.1.8 imposes additional requirements regarding RCS loop temperatures, THERMAL POWER and SDM requirements. The inclusion of these additional restrictions is considered reasonable to ensure operation within the bounds of the applicable safety analysis. The adoption of these requirements is an additional restriction on plant operation and is consistent with NUREG-1431.

M23 CTS 3.10.1.6 provides a SDM exception similar to that provided by ISTS 3.1.11. This SDM exception is not retained in the ITS since the measurement technique necessitating the SDM exception is no longer used. The elimination of the SDM exception is an additional restriction on plant operation and is consistent with NUREG-1431.

M24 CTS specifies measurement of control rod timing ". . . from the beginning of rod motion until dashpot entry." ITS specifies ". . . from the decay of stationary gripper coil voltages." The inclusion of the time from the beginning of stationary gripper coil voltage decay is an additional restriction on plant operation and is consistent with the NUREG-1431.

M25 CTS 3.10.6.3 establishes the action for one control rod inoperable to include changing the boron concentration to obtain an appropriate SDM. ITS 3.1.4 Action A increases this to one or more control rods inoperable and requires that either the SDM must be verified to be within limit or the boron concentration must be restored within the limit specified in the COLR, within one hour. Additionally, the plant must be placed in MODE 3 in 6 hours. Establishing a time limit (1 hour) to ensure the SDM requirement is satisfied is an additional restriction to current practices as is the new requirement to place the unit in MODE 3. Furthermore, since the ITS addresses one or more control rods inoperable and the CTS doesn't, the ITS requirement of placing the unit in a MODE outside the MODE of applicability within 6 hours is more restrictive than CTS 3.0 which allows 8 hours to be outside the MODE of applicability. These changes are more restrictive and are consistent with NUREG 1431.

M26 CTS 3.10.4.1 establishes the requirement for contol rod drop times but does not establish a related action if the drop times are not met. In this case Specification 3.0 would be entered. ITS 3.1.4 Condition includes control rod drop times not met and the associated Actions and Completion Times apply. These actions require that either the SDM must be verified to be within limit or the boron concentration must be restored within the limit specified in the COLR, within one hour. Additionally, the plant must be placed in MODE 3 in 6 hours. All of these actions are more restrictive than those found in CTS Specification 3.0 including the requirement to enter MODE 3 (outside the MODE of applicability) in 6 hours because the CTS 3.0 allows 8 hours to be outside the MODE of applicability. These changes are more restrictive and are consistent with NUREG 1431.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS Specifications 3.10.8.1, 3.10.8.2, and Figure 3.10-2 provide required shutdown margin values. CTS Specification 3.10.1.4 requires control rod insertion limits be adjusted to the end-of-core values as provided in the COLR at 50 percent of the cycle. These details are not retained in the ITS and are relocated to licensee controlled documents.

The details associated with the involved Specification are not required to be in the ITS to provide adequate protection of the public health and safety because the ITS still retains the requirement for compliance with the limits, and ITS Section 5.6 specifies the scope of the limits contained in the COLR and mandates NRC approval of the analytical methodology. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

LA2 CTS 3.10.1.3 requires that the reactor be placed in hot shutdown, and specifies that this be accomplished, "using normal operating procedures." This detail, specifying the manner in which to achieve hot shutdown, is relocated to licensee controlled documents.

The details associated with the involved Specification are not required to be in the ITS to provide adequate protection of the public health and safety because the ITS still retains the requirement for compliance with the Action. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 Since there is no action specified for failing to satisfy CTS 3.10.8.1 or 3.10.8.2, the required action is controlled by CTS 3.0. This CTS specification requires the unit be placed in hot shutdown within 8 hours followed by cold shutdown in an additional 30 hours. ITS 3.1.1 RA A.1, specifies initiating boration within 15 minutes to restore the SDM within limits. Both actions result in the addition of negative reactivity and a return to compliance with the assumptions of the safety analysis. ITS 3.1.1 RA A.1 requires timely restoration of SDM. Timely restoration of SDM is preferred to imposing the increased risk associated with a plant shutdown transient. Additionally, mandating shutdown of the unit may not be the safest course of action while sufficient SDM is not available. The proposed change provides an appropriate specific action for failing to satisfy the LCO instead of applying the generic action mandated by CTS 3.0. This change is consistent with NUREG-1431.

L2 This change involves two separate aspects both of which are analyzed separately here.

With the MTC outside the limits provided in the COLR, CTS 3.1.3.3 requires the reactor be made subcritical by an amount greater than or equal to the potential reactivity insertion due to depressurization. Since no completion time is explicitly stated, this specification implies completion as soon as practical. (Although not directly applicable, CTS 3.0 requires hot shutdown within 8 hours. Without a explicit statement of completion time, the comparable completion time in of 8 hours in CTS 3.0 is considered implicitly binding.) With MTC not within the upper limit, ITS 3.1.3 RA A.1 mandates establishment of administrative withdrawal limits for control banks to maintain MTC within the upper limit with a completion time of 24 hours. Provided ITS 3.1.3 RA A.1 is satisfied, no further action is required. While not explicitly stated, establishment of administrative withdrawal limits for control banks to maintain MTC within the upper limit is not precluded by CTS. However, the completion time of 24 hours to establish administrative control banks withdrawal limits is less restrictive than CTS permits.

With the required action or associated completion time of ITS 3.1.3 RA A.1 not met, ITS 3.1.3 RA B.1 mandates being in MODE 2 with $K_{eff} < 1.0$ within 6 hours. This completion time is in addition to the 24 hours permitted by ITS 3.1.3 RA A.1, and is less restrictive than CTS permits.

The proposed change provides a specific action and completion time for failing to satisfy the LCO. The completion time of 24 hours for ITS 3.1.1 RA A.1 provides sufficient time for evaluating the MTC measurement

and computing the required bank withdrawal limits. An action to permit addressing the specific condition is more suitable than an immediate plant shutdown, required by the CTS, with the associated increased risk for a shutdown transient.

With MTC outside the limits provided in the COLR, CTS 3.1.3.3 mandates being subcritical by an amount equal to the potential reactivity insertion due to depressurization. With MTC outside the upper limit, ITS 3.1.3 RA B.1 mandates, assuming ITS 3.1.3 RA A.1 and associated completion time not met, being in MODE 2 with $K_{eff} < 1.0$. In this condition, the SDM requirements of ITS LCO 3.1.1 are applicable requiring the SDM be within the limits provided in the COLR. The COLR includes appropriate SDM limits for this condition. Therefore this aspect of the change is administrative in nature.

L3 CTS Table 4.1-3, Item 2 requires verification of each control rods freedom of movement every 14 days during reactor critical operations. ITS SR 3.1.4.2 requires this surveillance to be performed at a 92 day Frequency and excludes control rods that are fully inserted. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because the 92 day Frequency takes into consideration other information available to the operator in the control room, and performance of SR 3.1.4.1, which verifies that individual rod positions are within alignment limits every 12 hours and adds to the determination of OPERABILITY of the rods. In addition, not requiring fully inserted rods to be exercised is less restrictive than the CTS which does not have this exception. This change is consistent with NUREG-1431.

L4 For control rod banks inserted in excess of the specified insertion limits, CTS 3.10.3 requires correction within one hour. ITS 3.1.6 RA A.2 permits two hours to restore the banks within limits. However, ITS also requires verification of SDM or initiation of boration to restore SDM within limits within one hour (see related DOC M17). Requiring the verification of SDM or the initiation of boration to restore SDM within one hour in concert with the restoration of control banks to within specified insertion limits within two hour provides some additional time to correct the condition while still restricting operation in this condition to a reasonably short time period. Prompt restoration of the control rod banks to within insertion limits is preferable to a plant shutdown with the associated risk of shutdown transients. This change is consistent with NUREG-1431.

L5 With the MTC outside the limits provided in the COLR, CTS 3.1.3.3 requires the reactor be made subcritical by an amount greater than or equal to the potential reactivity insertion due to depressurization. Since no completion time is explicitly stated, this specification

implies completion as soon as practical. (Although not directly applicable, CTS 3.0 requires hot shutdown within 8 hours. Without an explicit statement of completion time, the comparable completion time of 8 hours in CTS 3.0 is considered implicitly binding.) With MTC not within the lower limit, ITS 3.1.3 RA C.1 mandates being in MODE 4 with a completion time of 12 hours. This completion time is more than the implicit completion time for CTS 3.1.3.3. The requirement to be in MODE 4 is more restrictive than the CTS 3.1.3.3 requirement to be subcritical by an amount greater than or equal to the potential reactivity insertion due to depressurization. This change is considered a less restrictive change and is consistent with NUREG-1431.

RELOCATED SPECIFICATIONS

R1 3.10.7 Power Ramp Rate Limits

This Specifications, or Limiting Conditions for Operation (Chapter 3.0), is not retained in the ITS because it has been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provided in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of this Specification is provided in the report, "Application of Selection Criteria to the H. B. Robinson Steam Electric Plant Unit No. 2 Technical Specifications."

This Limiting Conditions for Operation, is relocated to licensee controlled documents. Relocation of the specific requirements for systems or variables contained in these Specifications to licensee documents will have no impact on the operability or maintenance of those systems or variables. The licensee will initially continue to meet the requirements contained in the relocated Specifications. The licensee is allowed to make changes to these requirements in accordance with the provisions of 10 CFR 50.59. Such changes can be made without prior NRC approval, if the change does not involve an unreviewed safety question, as defined in 10 CFR 50.59. These controls are considered adequate for assuring that structures, systems, and components in the relocated Specifications are maintained operable, and variables are maintained within limits. This change is consistent with the NRC Final Policy Statement on Technical Specification Improvements.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.1 - REACTIVITY CONTROL SYSTEMS

PART 3

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC), AND BASIS FOR CATEGORICAL EXCLUSION FROM 10 CFR 51.22

ADMINISTRATIVE CHANGES
("A" Labeled Comments/Discussions)

In the conversion of the HBRSEP Unit No. 2 Technical Specifications to the proposed plant specific Improved Technical Specifications certain wording preferences or conventions are being adopted which do not result in technical changes (either actual or interpretational). Editorial changes, clarification, reformatting, rewording and revised numbering are being adopted to make the improved Technical Specifications consistent with NUREG 1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Administrative" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes consist of editorial changes and clarification, reformatting, rewording and renumbering of the current Technical Specifications. This process does not involve any technical changes to existing requirements. As such, these changes are administrative in nature and do not impact initiators of analyzed events or alter any assumptions relative to mitigation of accident or transient events. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. The proposed changes do not impose or eliminate any requirements. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed changes will not reduce a margin of safety because they do not impact any safety analysis assumptions. These changes are administrative in nature and, as such, do not impact any technical requirements. Therefore, these changes do not involve any reduction in a margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
("M" Labeled Comments/Discussions)

The HBRSEP Unit No. 2 Technical Specifications are proposed to be modified in some areas to impose more restrictive requirements than currently exist. These more restrictive changes are being imposed to be consistent with NUREG 1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "More Restrictive" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes provide requirements determined to be more restrictive than the current Technical Specifications requirements for operation of the facility. These more restrictive requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. These changes have been confirmed to ensure that no previously evaluated accident has been adversely affected. The more restrictive requirements being proposed enhance assurance that process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis of the unit. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. These changes do impose new or additional requirements which are consistent with assumptions made in the safety analysis and licensing basis. The additional requirements include new Surveillance Requirements, more restrictive Frequencies and Completion Times, new LCOs, more restrictive Required Actions and Applicabilities, and other operational restrictions that enhance safe operation. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact or increases the margin of plant safety. Each of the changes in this category, while providing new or additional requirements designed to

enhance plant safety, is consistent with the safety analyses and licensing basis. Therefore, these changes do not involve a reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)
("LA" Labeled Comments/Discussions)

In the conversion of the HBRSEP Unit No. 2 Technical Specifications to the proposed plant specific Improved Technical Specifications, portions of some Specifications which are descriptive in nature regarding equipment, systems, actions, surveillances or programs are proposed to be relocated from the Specifications to the Bases, Updated Final Safety Analysis Report, procedures or other licensee controlled documents. The details associated with the involved specifications are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for compliance with the applicable specifications. Changes to the Bases are controlled in accordance with the proposed Bases Control Program described in Chapter 5 of the Improved Technical Specifications. Changes to the UFSAR and administrative procedures which control revisions to these relocated requirements are controlled in accordance with licensee controlled programs.

This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Less Restrictive-Generic" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes relocate requirements from the Technical Specifications to the Bases, Updated Final Safety Analysis Report, procedures or other licensee controlled documents. The documents containing the relocated requirements are subject to the change control of licensee controlled programs. Since any changes to these documents will be evaluated in accordance with the requirements of licensee controlled programs, no increase in the probability or consequences of an accident previously evaluated will be permitted without further NRC review. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. These changes do not introduce a new mode of

plant operation. Since any future changes to these requirements will be evaluated in accordance with licensee controlled programs, the possibility of a new or different kind of accident from any accident previously evaluated will not be permitted without further NRC review. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes will not reduce a margin of safety because they do not impact any safety analysis assumptions. The requirements that are transposed from the Technical Specifications to other licensee controlled documents are the same as the existing Technical Specifications. Since any future changes to these requirements will be evaluated in accordance with the requirements of licensee controlled programs, no reduction in any margin of safety will be permitted without further NRC review. Therefore, these changes do not involve any reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)
("L" Labeled Comments/Discussions)

L1 Change

Carolina Power & Light Company has evaluated the Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the change does not involve a significant hazards consideration are discussed below.

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

   The change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. Shutdown Margin (SDM) is not an initiator of analyzed events. However, SDM is an initial condition of analyzed events. As such, initiation of boration to restore SDM is preferred since the actions of CTS 3.0 will not result in prompt compliance with the safety analysis assumptions for SDM. Therefore, the change adds an action to initiate boration to restore SDM within 15 minutes. The probability of an event occurring during the ITS actions, while attempting to restore SDM, is no greater than the probability of an event occurring during the CTS actions. In addition, the consequences of an event occurring during the ITS actions, while attempting to restore SDM, are the same as the consequences of an event occurring during the CTS actions. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The change does provide an appropriate specific action for failing to satisfy the LCO instead of applying the generic action mandated by CTS 3.0. The change adds an action requiring the prompt restoration of SDM. Since dilution events have been previously analyzed, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   With SDM not within limits, the CTS does not provide a specific action. This necessitates entry into CTS 3.0 requiring hot shutdown in 8 hours followed by cold shutdown in an additional 30 hours. ITS 3.1.1, Required Action A.1 requires the initiation of boration within 15 minutes to restore the SDM to within limits. Provided that boration is initiated within 15 minutes, no shutdown is required.

Although not explicitly delineated in CTS, boration is still required to actually achieve the required SDM. Once the required SDM is achieved, CTS 3.0 can be exited. Hence, the only actual difference is that ITS does not mandate unit shutdown.

The margin of safety is not reduced in ITS by permitting boration to restore the SDM instead of requiring unit shutdown. Prompt restoration of SDM provides the greater margin to safety and is preferred to mandating unit shutdown. Therefore, there is no reduction in a margin of safety.

L2 Change

Carolina Power & Light Company has evaluated the Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the change does not involve a significant hazards consideration are discussed below.

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

   The change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. Moderator Temperature Coefficient (MTC) is not an initiator of an analyzed event. The probability of an event occurring during the ITS Actions, while establishing administrative controls to maintain MTC within limits, is no greater than the probability of an event occurring during the CTS actions. In addition, the consequences of an event occurring during the ITS Actions, while establishing administrative controls to maintain MTC within limits, are the same as the consequences of an event occurring during the CTS actions. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The change does provide an appropriate specific action for failing to satisfy the LCO instead of applying the generic action mandated by CTS 3.0. The change does not introduce a new mode of operation or changes in the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   A completion time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits. Permitting a reasonable time for actions to restore compliance with MTC limits, is

L4 Change

Carolina Power & Light Company has evaluated the Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the change does not involve a significant hazards consideration are discussed below.

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

   The change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. Insertion limits are not an initiator of analyzed events. The probability of an event occurring during the ITS actions, while restoring control rod insertion to within limits, is no greater than the probability of an event occurring during the CTS Actions. In addition, the consequences of an event occurring during the ITS Actions, while restoring control rod insertion to within limits, are the same as the consequences of an event occurring during the CTS actions. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The change provides a slight increase in the time to complete restoration of the control rod banks to within the insertion limits. Requiring the verification of SDM or the initiation of boration to restore SDM within one hour in concert with the restoration of control banks to within specified insertion limits within two hour provides some additional time to correct the condition while still restricting operation in this condition to a reasonably short time period. Prompt restoration of the control rod banks to within insertion limits is preferable to a plant shutdown with the associated risk of shutdown transients. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The change provides a slight increase in the time to complete restoration of the control rod banks to within the insertion limits. Requiring the verification of SDM or the initiation of boration to restore SDM within one hour in concert with the restoration of control banks to within specified insertion limits within two hours provides some additional time to correct the condition while still restricting operation in this condition to a reasonably short time period. Prompt restoration of the control rod banks to within insertion limits is preferable to a plant shutdown with the associated risk of shutdown transients. Therefore, this

change does not involve a significant reduction in a margin of safety.

L5 Change

Carolina Power & Light Company has evaluated the Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the change does not involve a significant hazards consideration are discussed below.

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

   The change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. Changing the completion time for entering MODE 4, for the MTC out of limits, from an implicit 8 hours in the CTS to 12 hours in the ITS does not increase the probability of occurrence of any analyzed event, since the function of the equipment, or limit for the parameter does not change. Further, the increase in time to place the plant in MODE 4 does not increase the consequences of an accident because a change in time to reduce power does not change the assumed response of equipment to perform its specific mitigation functions. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The change will still ensure compliance with the limiting condition for operation is maintained. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   There are no margins of safety related to any safety analysis that is dependent upon the change. The change increases the time allowed to place the plant in MODE 4 from an implicit 6 hours to 12 hours. Increasing the time to place the plant in MODE 4 when the MTC is out of limit provides additional time to place the plant in a condition outside the MODE of applicability in a controlled manner. Therefore, this change does not involve a significant reduction in a margin of safety.

RELOCATED CHANGES
("R" Labeled Comments/Discussions)

Relocating Requirements which do not meet the Technical Specification criteria to documents with an established control program allows the Technical Specifications to be reserved only for those conditions or limitations upon reactor operation which are necessary to adequately limit the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, thereby focusing the scope of Technical Specifications.

Therefore, requirements which do not meet the Technical Specification criteria in the NRC Final Policy Statement on Technical Specification Improvement for Nuclear Power Reactors (58FR 39132, dated 7/22/93) have been relocated to licensee controlled documents. This policy statement addresses the scope and purpose of Technical Specifications. In doing so, it establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria 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;

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;

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 barrier;

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

The application of these criteria is provided in the "Application of Screening Criteria to the HBRSEP Unit No. 2 Technical Specifications." Requirements which met the criteria have been included in the proposed improved Technical Specifications. Carolina Power & Light (CP&L) proposes to remove the requirements which do not meet the criteria from the Technical Specifications and relocate the requirements to a suitable owner controlled document. The requirements in the relocated Specifications are not affected by this Technical Specification change. CP&L will initially continue to perform the required operation and maintenance to assure that the requirements are satisfied. Relocating specific requirements for systems or variables has no impact on the system's operability or the variable's maintenance, as applicable.

Licensee controlled programs will be utilized as the control mechanism for the relocated Specifications as they will be placed in plant procedures or other licensee controlled documents. CP&L is allowed to make changes to these requirements, without prior NRC approval, if the change does not involve an unreviewed safety question. These controls are considered adequate for assuring structures, systems and components in the relocated Specifications are maintained operable and variables in the relocated Specifications are maintained within limits.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Relocated" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed change relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the "Application of Selection Criteria to the HBRSEP Unit No. 2 Technical Specifications." The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document under licensee control. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

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

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled document for which future changes will be evaluated pursuant to the requirements of licensee controlled programs. Therefore, this change does not involve a reduction in a margin of safety.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.1 - REACTIVITY CONTROL SYSTEMS

PART 4

MARKUP OF NUREG 1431, REVISION 1, "STANDARD TECHNICAL SPECIFICATIONS - WESTINGHOUSE PLANTS" (ISTS)

SDM ~~T_{avg} > 200°F~~ 3.1.1

3.1 REACTIVITY CONTROL SYSTEMS

CTS [3.10.8.1] [3.10.8.2]

3.1.1 SHUTDOWN MARGIN (SDM) ~~T_{avg} > 200°F~~

1

LCO 3.1.1 SDM shall be ~~≥ [1.6]% Δk/k~~.

2 Within the limits provided in the COLR

APPLICABILITY: MODE 2 with k_{eff} < 1.0.
MODES 3, 4, and 5.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
[L1]	A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

2

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
[M1]	SR 3.1.1.1 Verify SDM is within the limits provided in the COLR. ~~≥ [1.6]% Δk/k~~.	24 hours

ty

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 SHUTDOWN MARGIN (SDM) – T_{avg} ≤ 200°F

LCO 3.1.2 The SDM shall be ≥ [1.0]% Δk/k.

APPLICABILITY: MODE 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.2.1 Verify SDM is ≥ [1.0]% Δk/k.	24 hours

1

CTS
[4.9]

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	----------NOTE---------- The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. ---------- Verify measured core reactivity is within ± 1% Δk/k of predicted values.	Once prior to entering MODE 1 after each refueling AND -----NOTE------ Only required after 60 EFPD ---------- 31 EFPD thereafter

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Moderator Temperature Coefficient (MTC)

3

LCO 3.1.4 The MTC shall be maintained within the limits specified in the COLR. The maximum upper limit shall be ~~≤ [ ] Δk/k/°F at not zero power [that specified in Figure 3.1.4-1]~~ +5.0 pcm/°F at less than 50% RTP or 0.0 pcm/°F at 50% RTP and above

[3.1.3.1]

3

APPLICABILITY: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ for the upper MTC limit, MODES 1, 2, and 3 for the lower MTC limit.

[M4]

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[3.1.3.3] A. MTC not within upper limit.	A.1 Establish administrative withdrawal limits for control banks to maintain MTC within limit.	24 hours
[3.1.3.3] B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2 with $k_{eff} < 1.0$.	6 hours
[M5] C. MTC not within lower limit.	C.1 Be in MODE 4.	12 hours

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
[M 6]	SR 3.1.4.1 Verify MTC is within upper limit.	Once prior to entering MODE 1 after each refueling
TSTF-13	~~SR 3.1.4.2 Verify MTC is within 300 ppm Surveillance limit specified in the COLR.~~	~~----NOTE----~~ 1. Not required to be performed until 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm ~~Once each cycle~~
[M 6]	SR 3.1.4.3 ----------NOTES---------- If the MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR, SR 3.1.4.3 shall be repeated once per 14 EFPD during the remainder of the fuel cycle. SR 3.1.4.3 need not be repeated if the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR. ---------- Verify MTC is within lower limit.	~~----NOTE---- Not required to be performed until 7 EFPD after reaching the equivalent of an equilibrium RTP-ARO boron concentration of 300 ppm~~ TSTF-13 Once each cycle

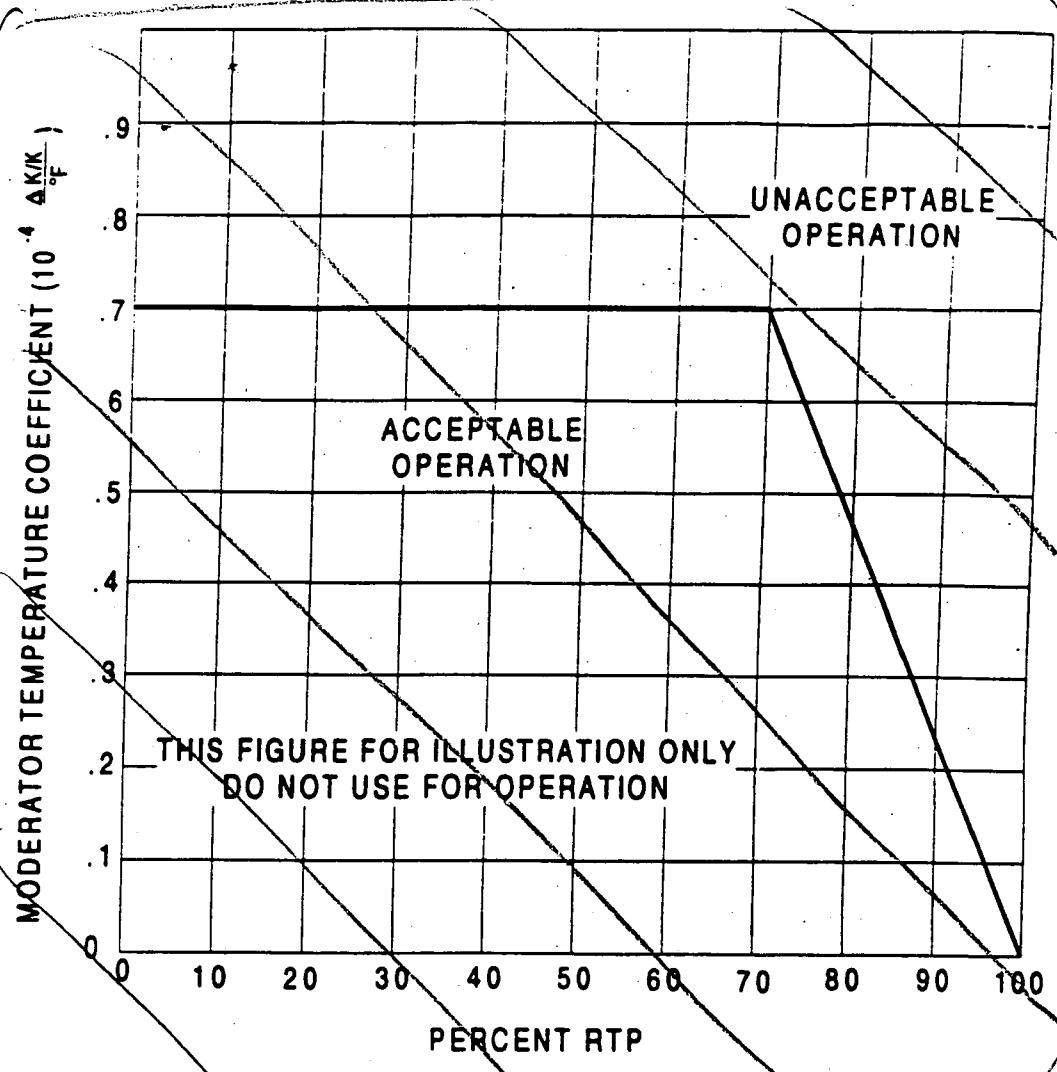

Figure 3.1.4-1 (page 1 of 1)
Moderator Temperature Coefficient vs. Power Level

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Rod Group Alignment Limits

[3.10.6.1]

LCO 3.1.5 All shutdown and control rods shall be OPERABLE. with all individual indicated rod positions ~~within 12 steps of their group step counter demand position~~.

[3.10.1.5]

AND

shall be

Insert 3.1.4-1

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[CTS 3.10.6.3] A. One or more rod(s) untrippable. (Inoperable)	A.1.1 Verify SDM is ~~≥ [1.6]% Δk/k~~ OR within the limits provided in the COLR	1 hour
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
[M9]	A.2 Be in MODE 3.	6 hours
[3.10.1.5] B. One rod not within alignment limits.	B.1 Restore rod to within alignment limits.	1 hour
	OR	
[M9]	B.2.1.1 Verify SDM is ~~≥ [1.6]% Δk/k~~ OR within the limits provided in the COLR	1 hour

(continued)

ITS Insert 3.1.4-1 (Rod Group Alignment Limits)

as follows:

a. For bank demand positions ≥ 200 steps, each rod shall be within 15 inches of its bank demand position, and

b. For bank demand positions < 200 steps, each rod shall be within 7.5 inches of the average of the individual rod positions in the bank.

CTS

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
[M 9]	B. (continued)	B.2.1.2 Initiate boration to restore SDM to within limit.	1 hour
		AND	
[3.10.1.5]		B.2.2 Reduce THERMAL POWER to ≤ ~~75%~~ 70% RTP. (6)	2 hours
		AND	
[M 9]		B.2.3 Verify SDM is ~~≤ [1.6]% Δk/k ...~~ within the limits provided in the COLR (2)	Once per 12 hours
		AND	
[3.10.1.5.b]		B.2.4 Perform SR 3.2.1.1.	72 hours
		AND	
[3.10.1.5.b]		B.2.5 Perform SR 3.2.2.1.	72 hours
		AND	
[M 9]		B.2.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days
[M 9]	C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
[M9] D. More than one rod not within alignment limit.	D.1.1 Verify SDM is ~~[illegible]~~ within the limits provided in the COLR	1 hour
	OR	
[M9]	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	AND	
[M9]	D.2 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
[M12] SR 3.1.5.1 Verify individual rod positions within alignment limit.	12 hours AND Once within 4 hours and every 4 hours thereafter when the rod position deviation monitor is inoperable

(continued)

SURVEILLANCE REQUIREMENTS (continued)

CTS

SURVEILLANCE		FREQUENCY
SR 3.1.~~5~~ 4.2 [Table 4.1-3 Item 2]	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	92 days
SR 3.1.~~5~~ 4.3 [Table 4.1-3 Item 1] [3.10.4.1]	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ ~~[2.2]~~ 1.8 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ~~≥ 500°F~~ ≥ 540°F; and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head 7

3.1 REACTIVITY CONTROL SYSTEMS

CTS

3.1.6 Shutdown Bank Insertion Limits
5

[3.10.1.2] LCO 3.1.6 Each shutdown bank shall be within insertion limits specified in the COLR.
5

APPLICABILITY: MODE 1.
MODE 2 with any control bank not fully inserted.

[3.10.1.6] ----------NOTE----------
This LCO is not applicable while performing SR 3.1.6.2.
4

8

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[M 14] A. One or more shutdown banks not within limits. (both)	A.1.1 Verify SDM is ≥ [1.6]% Δk/k within the limits provided in the COLR. 3	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
[M 14]	A.2 Restore shutdown banks to within limits.	2 hours
[M 14] B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
[M15] SR 3.1.6.1 5	Verify each shutdown bank is within the limits specified in the COLR.	12 hours

CTS

3.1 REACTIVITY CONTROL SYSTEMS

[3.10.1] 3.1 7 Control Bank Insertion Limits
6

[3.10.1.3] LCO 3.1 8 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.
6

APPLICABILITY: MODE 1.
MODE 2 with $k_{eff} \geq 1.0$.

[3.10.1.6]

----------NOTE----------
This LCO is not applicable while performing SR 3.1 8.2.
4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[M17] A. Control bank insertion limits not met.	A.1.1 Verify SDM is ≥ 1.6% Δk/k within the limits provided in the COLR	1 hour 2
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
[3.10.1.3]	A.2 Restore control bank(s) to within limits.	2 hours

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
[M17] B. Control bank sequence or overlap limits not met.	B.1.1 Verify SDM is ≥1.6% Δk/k within the limits provided in the COLR. OR	1 hour
	B.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
[M17]	B.2 Restore control bank sequence and overlap to within limits.	2 hours
[3.10.1.3] C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours

2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
[M18] SR 3.1.7.1 6	Verify estimated critical control bank position is within the limits specified in the COLR.	Within 4 hours prior to achieving criticality

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
[M1B]	SR 3.1.7.2 6	Verify each control bank insertion is within the limits specified in the COLR.	12 hours AND Once within 4 hours and every 4 hours thereafter when the rod insertion limit monitor is inoperable
[M1B]	SR 3.1.7.3 6	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	12 hours

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Rod Position Indication
7

[M19] LCO 3.1.8 The [Digital] Analog Rod Position Indication ([D]A RPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
[M19] Separate Condition entry is allowed for each inoperable rod position indicator per group and each demand position indicator per bank.

CONDITION	REQUIRED ACTION		COMPLETION TIME
[M19] A. One [D]A RPI per group inoperable for one or more groups.	A.1	Verify the position of the rods with inoperable position indicators by using movable incore detectors.	Once per 8 hours
	OR		
	A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
[M19] B. One or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	B.1	Verify the position of the rods with inoperable position indicators by using movable incore detectors.	~~24~~ hours 6 9
	OR		
			(continued)

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[M17] B. (continued)	B.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
[M19] C. One demand position indicator per bank inoperable for one or more banks.	C.1.1 Verify by administrative means all ~~DI~~RPIs for the affected banks are OPERABLE. (A)	Once per 8 hours
	AND	
Verify the position of each rod in the affected bank(s) is within 7.5 inches of the average of the individual rod positions in the affected bank(s).	C.1.2 ~~Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart.~~ (10)	Once per 8 hours
	OR	
[M19]	C.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
[M19] D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
[M19] SR 3.1.8.1	Verify each ~~[D]~~ A RPI agrees within ~~[12] steps~~ of the group demand position for the full indicated range of rod travel.	~~[18 months]~~ 10 11

the requirements established in LCO 3.1.4

Once prior to Criticality after each removal of the reactor vessel head.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.9 PHYSICS TESTS Exceptions – MODE 1

LCO 3.1.9 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.5, "Rod Group Alignment Limits";
LCO 3.1.6, "Shutdown Bank Insertion Limits";
LCO 3.1.7, "Control Bank Insertion Limits";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and
LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)"

may be suspended, provided:

a. THERMAL POWER is maintained ≤ 85% RTP;

b. Power Range Neutron Flux – High trip setpoints are ≤ 10% RTP above the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP; and

c. SDM is ≥ [1.6]% Δk/k.

APPLICABILITY: MODE 1 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. SDM not within limit.	A.1	Initiate boration to restore SDM to within limit.	15 minutes
	AND		
	A.2	Suspend PHYSICS TESTS exceptions.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. THERMAL POWER not within limit.	B.1 Reduce THERMAL POWER to within limit.	1 hour
	OR	
	B.2 Suspend PHYSICS TESTS exceptions.	1 hour
C. Power Range Neutron Flux – High trip setpoints > 10% RTP above the PHYSICS TEST power level. OR Power Range Neutron Flux – High trip setpoints > 90% RTP.	C.1 Restore Power Range Neutron Flux – High trip setpoints to ≤ 10% above the PHYSICS TEST power level, or to ≤ 90% RTP, whichever is lower.	1 hour
	OR	
	C.2 Suspend PHYSICS TESTS exceptions.	1 hour

12

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Verify THERMAL POWER is ≤ 85% RTP.	1 hour
SR 3.1.9.2	Verify Power Range Neutron Flux – High trip setpoints are ≤ 10% above the PHYSICS TEST power level, and ≤ 90% RTP.	Within 8 hours prior to initiation of PHYSICS TESTS
SR 3.1.9.3	Perform SR 3.2.1.1 and SR 3.2.2.1.	12 hours
SR 3.1.9.4	Verify SDM is ≥ [1.6]% Δk/k.	24 hours

12

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.~~10~~ 8 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.~~10~~ 8 During the performance of PHYSICS TESTS, the requirements of

[3.1.3.1] LCO 3.1.~~4~~ 3 "Moderator Temperature Coefficient (MTC)";
[3.10.1.5] LCO 3.1.~~5~~ 4 "Rod Group Alignment Limits";
[3.10.1.2] LCO 3.1.~~6~~ 5 "Shutdown Bank Insertion Limits";
[3.10.1.3] LCO 3.1.~~7~~ 6 "Control Bank Insertion Limits"; and
[3.1.3.1] LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended, provided:

[M22] a. RCS lowest loop average temperature is ≥ ~~[551]~~ 530°F; and

[3.10.1.6] b. SDM is ~~≥ [1.6]% Δk/k~~ within limits provided in the COLR; and 2

C. THERMAL POWER is < 5% RTP

APPLICABILITY: MODE 2 during PHYSICS TESTS. TSTF-14

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
[M20] A. SDM not within limit.	A.1	Initiate boration to restore SDM to within limit.	15 minutes
	AND		
	A.2	Suspend PHYSICS TESTS exceptions.	1 hour
[M20] B. THERMAL POWER not within limit.	B.1	Open reactor trip breakers.	Immediately

(continued)

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
[M20]	C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes
[M20]	D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
	~~SR 3.1.10.1~~	~~Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per [SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1].~~	~~Within 12 hours prior to initiation of PHYSICS TESTS~~ 13
[M21]	SR 3.1.~~10~~ 8.8	Verify the RCS lowest loop average temperature is ≥ ~~[531]~~ 530 °F.	30 minutes
[M21]	SR 3.1.~~10~~ 8.3	Verify SDM is ~~≥ 1.6% Δk/k~~ within limits provided in COLR	24 hours 2
[M21]	SR 3.1.8.2	VERIFY THERMAL POWER IS ≤ 5% RTP,	30 minutes

TSTF-14

12

3.1 REACTIVITY CONTROL SYSTEMS

3.1.11 SHUTDOWN MARGIN (SDM) Test Exceptions

LCO 3.1.11 The SDM requirements in MODE 2 may be suspended, provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2 when measuring control rod worth and SDM.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more control rods not fully inserted. AND Available trip reactivity from OPERABLE control rods less than the highest estimated control rod worth.	A.1 Initiate boration to restore SDM to within limit.	15 minutes
B. All control rods fully inserted. AND Reactor subcritical by less than the highest estimated control rod worth.	B.1 Initiate boration to restore SDM to within limits.	15 minutes

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.11.1	----------NOTE---------- Only required for control rods not fully inserted. ---------- Determine the position of each control rod.	2 hours
SR 3.1.11.2	----------NOTE---------- Only required for control rods not fully inserted. ---------- Trip each control rod from ≥ the 50% withdrawn position, and verify full control rod insertion.	Within 24 hours prior to reducing SDM outside limits

12

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.1 - REACTIVITY CONTROL SYSTEMS

PART 5

JUSTIFICATION FOR DIFFERENCES (JFDs) TO ISTS

1 ISTS Specification 3.1.2 is not included as a separate specification in the ITS. Since the specific shutdown margin requirements for various plant conditions are relocated to the Core Operating Limits Report (COLR), there is no need for separate specifications for different MODES of Applicability. Consequently, shutdown margin requirements applicable to MODE 5 are included in ITS Specification 3.1.1. This eliminates the need for Specification 3.1.2. Subsequent Specifications are renumbered accordingly.

2 Specific numerical values for SDM located throughout the Technical Specifications have been relocated to the COLR. SDM is a cycle-specific variable similar to Moderator Temperature Coefficient, Rod insertion Limits, Axial Flux Difference, Heat Flux Hot Channel Factor, and Nuclear Enthalpy Rise Hot Channel Factor, which are currently contained in the COLR. In addition, there is an NRC-approved methodology for calculating SDM. Relocating SDM to the COLR provides core design and operational flexibility that can be used for improved fuel management and to solve plant specific issues. If the SDM is retained in the COLR the core design can be finalized after shutdown, when the actual end of cycle burnup is known. This can save redesign efforts if the actual burnup differs from the projected value. Currently, reload design efforts and resolution of plant specific issues are somewhat restricted, since a change in the SDM requires a License Amendment.

3 ISTS Figure 3.1.4-1 is not used in the ITS. The maximum upper limit for MTC consists of two values specified in ITS 3.1.3, obviating any need for the Figure.

4 ITS Specification 3.1.5, "Rod Group Alignment Limits," consists of two separate requirements: 1) shutdown and control for rod OPERABILITY (defined in the Bases as the ability to insert on an RPS trip), and 2) indicated position of each rod within 12 steps of its group demand position (i.e., correctly positioned). These requirements have been separated in the LCO and Actions to ensure the appropriate actions are taken for each condition. Condition A wording is broadened from "untrippable" to "inoperable" such that the condition encompasses all rod inoperability conditions. Without this change, it is ambiguous with regard to a rod with a slow drop time but one that is still trippable.

5 ITS 3.1.4 rod group alignment limits are modified to be consistent with current licensing basis.

6 ITS 3.1.4, Required Action B.2.2, which requires reducing THERMAL POWER to ≤ 75 percent RTP, is modified to specify reducing THERMAL POWER to ≤70% RTP, consistent with current licensing basis.

7 ITS SR 3.1.4.3 is modified to reflect a minimum T_{avg} of 540°F for verification of rod drop times, consistent with current licensing basis.

8 The word, "more," is changed to the word, "both," because plant design includes two shutdown banks.

9 ITS Specification 3.1.7, Required Action B.1, requires verification of rod position using the movable incore detectors for rods with inoperable position indications that have moved in excess of 24 steps since the last determination of rod position. The bracketed Completion Time of 4 hours is modified to 6 hours. Since the CTS does not include a comparable requirement, there is no current licensing basis for this value. A Completion Time of 6 hours is considered to be a reasonable time in which to perform the required flux mapping and data analysis. A Completion Time of 6 hours still provides sufficient time to complete alternate Required Action B.2, reduction of THERMAL POWER to ≤ 50% RTP within 8 hours.

10 ITS Specification 3.1.7, Required Action C.1.2, is modified to be consistent with ITS Specification 3.1.4, as modified, and current licensing basis.

11 ITS SR 3.1.7.1 Frequency is adopted such that this SR is performed any time the reactor vessel head is removed. By not specifying a fixed Frequency, any reactor vessel head removal requires that the SR be performed to verify OPERABILITY of the Rod Position Indicator System.

12 ISTS Specification 3.1.9, "PHYSICS TEST Exceptions - MODE 1," is not adopted in the ITS. These physics tests are not performed during post-refueling startup testing. ISTS Specification 3.1.11, "SDM Test Exceptions," is not adopted in the ITS. The use of other rod worth measurement techniques will maintain the shutdown margin during the entire measurement process and still provide the necessary physics data verification. Since the N-1 measurement technique is no longer used, the SDM test exception is not necessary. Subsequent Specifications are renumbered accordingly.

13 ITS SR 3.1.8.1 is deleted, and subsequent SRs are renumbered accordingly. Performance of a COT on power range and intermediate range channels is required by ITS LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," every 92 days (SR 3.3.1.7 and SR 3.3.1.8). The 92 day required Frequency has been determined to be sufficient for verification that the power range and intermediate range monitors are properly functioning.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.1 - REACTIVITY CONTROL SYSTEMS

PART 6

MARKUP OF ISTS BASES

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM) (T_{avg} > 200°F)

BASES

HBRSEP Design Criteria

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod CUSTED assembly of highest reactivity worth is fully withdrawn and the fuel and moderator temperatures are changed to the normal hot zero power value

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Control Rod System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Control Rod System, together with the boration system provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.8, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

(continued)

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1

BASES (continued)

APPLICABLE SAFETY ANALYSES

the

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out (QN) scram.

Insert B3.1.1-1

8

following a reactor

the

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 280 cal/gm energy deposition for the rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases ~~until the MODE 5 value is reached~~. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to

9

(continued)

ITS INSERT B3.1.1-1

For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

APPLICABLE SAFETY ANALYSES (continued)

power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition;

c. Startup of an inactive reactor coolant pump (RCP); and

d. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor

(continued)

APPLICABLE SAFETY ANALYSES (continued)

coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 2 with k_{eff} < 1.0 and in MODES 3 and 4 and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. [In MODE 5, SDM is addressed by LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - T_{avg} ≤ 200°F."] In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.6, "Shutdown Bank Insertion Limits," and LCO 3.1.5.

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that

(continued)

BASES

ACTIONS

A.1 (continued)

boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank, or the borated water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% Δk/k must be recovered and a boration flow rate of [ ] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% Δk/k. These boration parameters of [ ] gpm and [ ] ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.6 and LCO 3.1.7 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

In MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation considering the listed reactivity effects:

a. RCS boron concentration;

b. Control bank position;

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

c. RCS average temperature; previous critical boron concentration

d. Fuel burnup based on ~~gross thermal energy generation~~

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the ~~calculation~~ verification

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 26~~ UFSAR, Section 3.1
2. UFSAR, ~~Chapter [15]~~ Section 15.1.5
3. UFSAR, ~~Chapter [15]~~ Section 15.4.6
4. 10 CFR 100.
5. UFSAR, TABLE 15.4.6-1
6. UFSAR, TABLE 9.3.4-1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 SHUTDOWN MARGIN (SDM) – T_{avg} ≤ 200°F

BASES

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming the single rod cluster assembly of highest reactivity worth is fully withdrawn. (3)

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Control Rod System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Control Rod System, together with the boration system, provides SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.7, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in the safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs with the assumption of the highest worth rod stuck out on scram. Specifically, for MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio, fuel centerline temperature limits for AOOs, and ≤ 280 cal/gm energy deposition for the rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life when critical boron concentrations are highest.

SDM satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

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(continued)

BASES (continued)

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The boron dilution accident (Ref. 2) is the most limiting analysis that establishes the SDM value of the LCO. For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 2, with k_{eff} ≥ 1.0 and MODES 3 and 4, the SDM requirements are given in LCO 3.1.1, "SHUTDOWN MARGIN (SDM) – T_{avg} > 200°F." In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODE 1 and MODE 2, with k_{eff} ≥ 1.0, SDM is ensured by complying with LCO 3.1.6, "Shutdown Bank Insertion Limits," and LCO 3.1.7.

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

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In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank or the borated water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate the time in core life must be considered. For instance, the most difficult time

(continued)

BASES

ACTIONS

A.1 (continued)

in core life to increase the RCS boron concentration is at the beginning of cycle, when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% Δk/k must be recovered and a boration flow rate of [ ] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% Δk/k. These boration parameters of [ ] gpm and [ ] ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

In MODE 5, the SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM. This allows time enough for the operator to collect

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Chapter [15].

3

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 Core Reactivity

HBRSEP Design Criteria

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM) (k_{eff} < 0.99 SDM") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the

(continued)

BACKGROUND (continued)

calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity

(continued)

APPLICABLE SAFETY ANALYSES (continued)

behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of the NRC Policy Statement.

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of ± 1% $\Delta k/k$ has been established based on engineering judgment. A 1% deviation in reactivity from

(continued)

LCO (continued)

that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1% Δk/k of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. An SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of

(continued)

ACTIONS

A.1 and A.2 (continued)

the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 72 hours is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then the boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by a Note. The Note indicates that the normalization of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly.

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.~~
2. FSAR, Chapter 15.

UFSAR Section 3.1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Moderator Temperature Coefficient (MTC)

BASES

HBRSEP Design Criteria

BACKGROUND

Insert B 3.1.3-1

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than or equal to zero when THERMAL POWER is at RTP. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the FSAR accident and transient analyses.

(continued)

ITS INSERT B3.1.3-1

the reactor core with its related controls and protection systems are designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations.

BACKGROUND (continued)

If the LCO limits are not met, the unit response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The FSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from either zero (Ref. 4) or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

whether the reactor is at full or zero power, and whether it is the BOC or EOC life. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. An EOC measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

MTC satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.4 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the ~~original~~ accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at BOC; this upper bound must not be exceeded. This maximum upper limit occurs at BOC, all rods out (ARO), hot zero power conditions. At EOC the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

The LCO establishes a maximum positive value that cannot be exceeded. The BOC positive limit and the EOC negative limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take

(continued)

LCO (continued)

advantage of improved fuel management and changes in unit operating schedule.

APPLICABILITY

Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled CONTROL ROD assembly or group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

If the BOC MTC limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits.

As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

(continued)

ACTIONS (continued)

B.1

If the required administrative withdrawal limits at BOC are not established within 24 hours, the unit must be brought to MODE 2 with $k_{eff} < 1.0$ to prevent operation with an MTC that is more positive than that assumed in safety analyses.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

C.1

Exceeding the EOC MTC limit means that the safety analysis assumptions for the EOC accidents that use a bounding negative MTC value may be invalid. If the EOC MTC limit is exceeded, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 4 within 12 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.~~4~~ 3 1 3

This SR requires measurement of the MTC at BOC prior to entering MODE 1 in order to demonstrate compliance with the most positive MTC LCO. Meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOC MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the BOC MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2 and SR 3.1.4.3 — TSTF-13

In similar fashion, the LCO demands that the MTC be less negative than the specified value for EOC full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the EOC LCO limit. The 300 ppm SR value is sufficiently less negative than the EOC LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met.

assumed in the most limiting accident analysis

SR 3.1.4.3 is modified by a Note that includes the following requirements:

TSTF13

b. If the 300 ppm Surveillance limit is exceeded, it is possible that the EOC limit on MTC could be reached before the planned EOC. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the EOC limit.

c. The Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is more positive than the 60 ppm Surveillance limit, the EOC limit will not be exceeded because of the gradual manner in which MTC changes with core burnup.

less negative

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11 — UFSAR Section 3.1
2. UFSAR, Chapter [15] — 15.0.5
3. WCAP 9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
4. UFSAR, Chapter [15] — 15.4.1 — TSTF-13

a. The SR is not required to be performed until 7 effective full power days (EFPDs) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

Insert B 3.1.4-1

~~The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).~~

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately ⅝ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups

(continued)

ITS INSERT B3.1.4-1

The applicable criteria for these reactivity and power distribution design requirements are described in the UFSAR (Ref. 1) and 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref 2).

BASES

HBRSEP has

BACKGROUND (continued)

that are moved in a staggered fashion, but always within one step of each other. All units have four control banks and at least two shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

for the remaining control banks.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

Analog

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ⅝ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is six steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the DRPI will go on half

(continued)

BASES

BACKGROUND (continued)

~~accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is ± 6 steps (± 3.75 inches) and the~~ The maximum uncertainty of the ARPI System is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and ARPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches. (Ref. 4 and 6)

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

~~a. There be no violations of:~~

a. 1. specified acceptable fuel design limits, or

b. 2. Reactor Coolant System (RCS) pressure boundary integrity. ~~and~~

~~b. The core remains subcritical after accident transients.~~

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Two types of analysis are performed in regard to static rod misalignment (Ref. 4). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted in excess of its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 5).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of the NRC Policy Statement.

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements also ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

(continued)

ITS INSERT B3.1.4-2

The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required time on a valid signal. CRDM malfunctions that result in inability to move a rod (e.g., rod urgent failures), which do not impact trippability, do not necessarily result in rod inoperability.

BASES

LCO (continued)

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are bottomed and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM) (T_{avg} > 200°F)" for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

ACTIONS

A.1.1 and A.1.2

When one or more rods are untrippable, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM.

In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

A.2

If the untrippable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve

(continued)

BASES

ACTIONS

A.2 (continued)

this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1

When a rod becomes misaligned, it can usually be moved and is still trippable. If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.6, "Shutdown Bank Insertion Limits," and LCO 3.1.7, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be moved fully in and control bank C must be moved in to approximately 100 to 115 steps.

Power operation may continue with one RCCA trippable but misaligned, provided that SDM is verified within 1 hour.

(continued)

ACTIONS

B.2.1.1 and B.2.1.2 (continued)

The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 7). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

(continued)

BASES

ACTIONS (continued)

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

E.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power

(continued)

BASES

ACTIONS

A.1 (continued)

conditions in an orderly manner and without challenging the plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. If the rod position deviation monitor is inoperable, a Frequency of 4 hours accomplishes the same goal. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

SR 3.1.5.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.5.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.5.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable and aligned, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

by the Normal CRDM

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.5.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions.

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. FSAR, Chapter [15].
4. FSAR, Chapter [15].
5. FSAR, Chapter [15].
6. FSAR, Chapter [15].
7. FSAR, Chapter [15].

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

Insert B3.1.5-1

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. All HBRSEP plants have four control banks and at least two shutdown banks. See LCO 3.1.5, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.8, "Rod Position Indication," for position indication requirements.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations.

(continued)

ITS INSERT B3.1.5-1

The applicable criteria for these reactivity and power distribution design requirements are described in the UFSAR (Ref. 1) and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref 2).

BASES

BACKGROUND (continued)

Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown banks are then left in this position until the reactor is shut down. They affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.5, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM) - T_{avg} > 200°F," and LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - T_{avg} ≤ 200°F") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

There be no violations of:

a. specified acceptable fuel design limits, or

b. RCS pressure boundary integrity, and

~~b. The core remains subcritical after accident transients.~~

As such, the shutdown bank insertion limits affect safety analysis involving core reactivity and SDM (Ref. 3).

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of the NRC Policy Statement.

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

APPLICABILITY

The shutdown banks must be within their insertion limits with the reactor in MODES 1 and 2. The applicability in MODE 2 begins prior to initial control bank withdrawal during an approach to criticality, and continues throughout MODE 2, until all control bank rods are again fully inserted by reactor trip or by shutdown. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6, the shutdown banks ~~are fully inserted in the core~~ and contribute to the SDM. Refer to LCO 3.1.1 ~~and LCO 3.1.2~~ for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.5.2. This SR verifies the freedom of the rods to

(continued)

BASES

APPLICABILITY (continued)

move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1.1, A.1.2 and A.2

When one or more shutdown banks is not within insertion limits, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1 (continued)

shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28~~ UFSAR, Section 3.1
2. 10 CFR 50.46.
3. FSAR, Chapter 15

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

INSERT 3.1.6-1

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. Plants have four control banks and at least two shutdown banks. See LCO 3.1.8, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.8, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. An example is provided for information only in Figure B 3.1.7-1. The control banks are required to be at or above the insertion limit lines.

Figure B 3.1.7-1 also indicates how the control banks are moved in an overlap pattern. Overlap is the distance travelled together by two control banks. The predetermined

(continued)

ITS INSERT B3.1.6-1

The applicable criteria for these reactivity and power distribution design requirements are described in the UFSAR (Ref. 1) and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref 2).

BASES

BACKGROUND (continued)

position of control bank C, at which control bank D will begin to move with bank C on a withdrawal, will be at 128 steps for a fully withdrawn position of 225 steps. The fully withdrawn position is defined in the COLR.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Rod Control System, but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.1.5, LCO 3.1.6, "Shutdown Bank Insertion Limits," LCO 3.1.7, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

a. There be no violations of:

1. specified acceptable fuel design limits, or
2. Reactor Coolant System pressure boundary integrity; and

b. The core remains subcritical after accident transients.

As such, the shutdown and control bank insertion limits affect safety analysis involving core reactivity and power distributions (Ref. 3).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 4).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 5).

The insertion limits satisfy Criterion 2 of the NRC Policy Statement, in that they are initial conditions assumed in the safety analysis.

LCO

The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate

(continued)

BASES

LCO (continued) - negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

APPLICABILITY The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

The applicability requirements have been modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.5.2. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO.

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2

When the control banks are outside the acceptable insertion limits, they must be restored to within those limits. This restoration can occur in two ways:

a. Reducing power to be consistent with rod position; or

b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ~~(T_{avg} > 200°F)~~") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

(continued)

BASES

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2 (continued)

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlaps limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

C.1

If the Required Actions ~~A.1 and A.2 or B.1 and B.2~~ cannot be completed within the associated Completion Times, the plant must be brought to MODE 3, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1 (continued)

that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.7.2

With an OPERABLE bank insertion limit monitor, verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY of the bank insertion limit monitor and to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours. If the insertion limit monitor becomes inoperable, verification of the control bank position at a Frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits.

SR 3.1.7.3

When control banks are maintained within their insertion limits as checked by SR 3.1.7.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.7.2.

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28.~~ UFSAR Section 3.1

2. 10 CFR 50.46.

3. FSAR, Chapter [15]

~~4. FSAR, Chapter [15].~~

~~5. FSAR, Chapter [15]~~

Control Bank Insertion Limits
B3.1.6

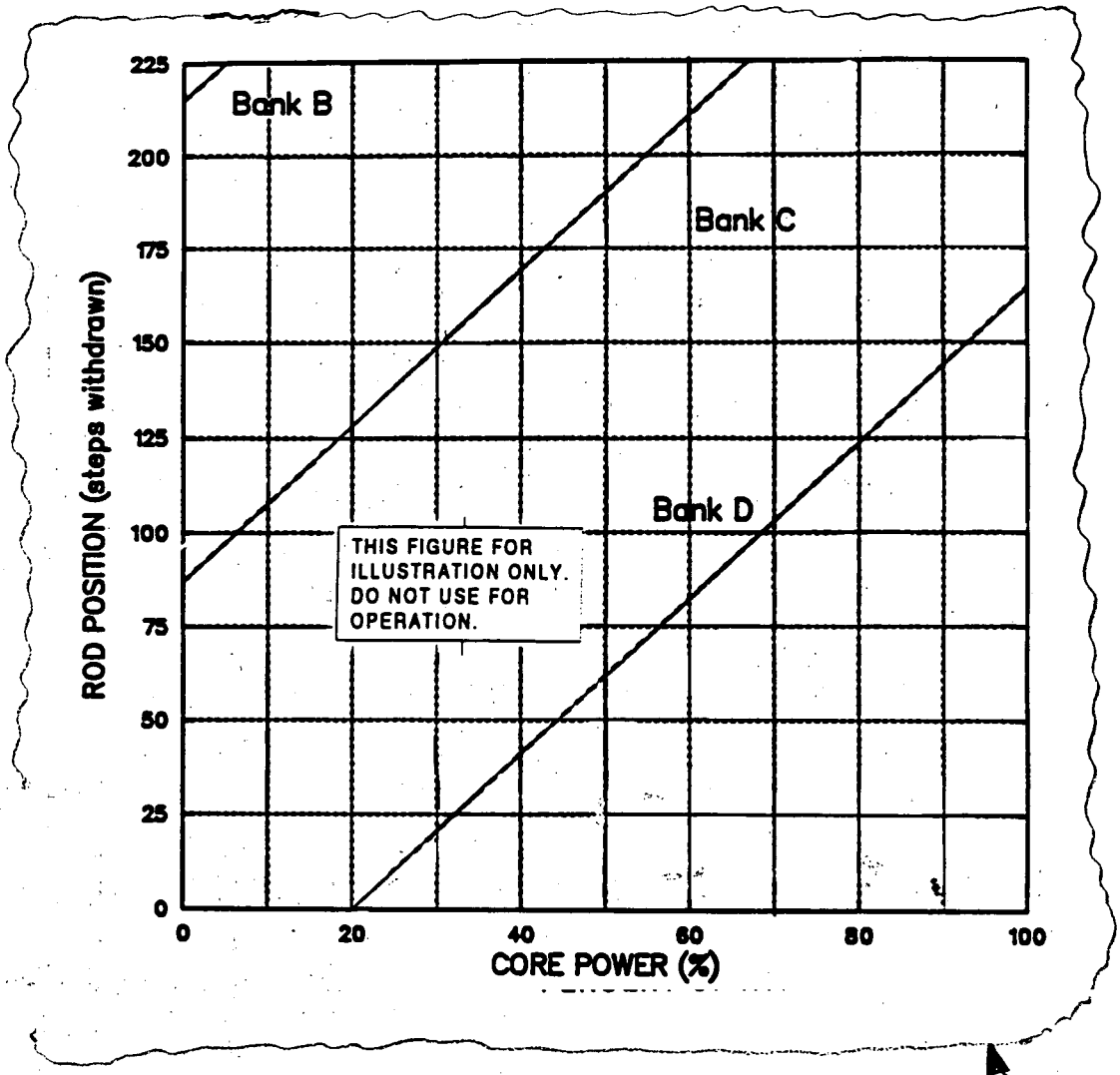

Figure B 3.1.6-1 (page 1 of 1)
Control Bank Insertion vs. Percent RTP

B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1.8 Rod Position Indication

BASES

BACKGROUND

Insert B 3.1.7-1

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.8 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control.

(continued)

ITS INSERT B3.1.7-1

The applicable design criteria for rod position indication are described in the UFSAR (Ref. 1).

BASES

BACKGROUND (continued)

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ⅝ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is 6 steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the DRPI will go on half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.6, "Shutdown Bank Insertion Limits," and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.1.7 "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.5 "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of the NRC Policy Statement. The control rod position indicators monitor control rod position, which is an initial condition of the accident.

LCO

LCO 3.1.8 specifies that one DRPI System and one Bank Demand Position Indication System be OPERABLE for each control rod. For the control rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following:

a. The DRPI System ~~indicates within 12 steps of the group step counter demand position as required by~~ Meets the requirements of LCO 3.1.5 "Rod Group Alignment Limits";

b. For the DRPI System there are no KNOWN failed coils; and

c. The Bank Demand Indication System ~~has been calibrated either in the fully inserted position or to the DRPI System~~ had been previously reset to zero with all rods

~~The 12 step agreement limit between the Bank Demand Position Indication System and the DRPI System indicates that the Bank Demand Position Indication System is adequate, calibrated, and~~ By meeting the requirements of LCO 3.1.4 can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.5, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

(continued)

BASES

LCO (continued)

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements on the RPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.5, LCO 3.1.6, and LCO 3.1.7), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator per group and each demand position indicator per bank. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1

When one RPI channel per group fails, the position of the rod can still be determined by use of the incore movable detectors. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of B.1 or B.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

(continued)

BASES

ACTIONS (continued)

A.2

Reduction of THERMAL POWER to ≤ 50% RTP ~~puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 3)~~ more than offsets the increases in core F_Q and $F^N_{\Delta H}$ due to rod position

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1 and B.2

These Required Actions clarify that when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 are still appropriate but must be initiated promptly under Required Action B.1 to begin verifying that these rods are still properly positioned, relative to their group positions.

If, within 24 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 24 hours provides an acceptable period of time to verify the rod positions.

C.1.1 and C.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the DRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and ~~the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart~~ that the position of each rod in the affected bank(s) is within 7.5 inches of the average of the individual rod positions in the affected bank(s) within the allowed Completion Time of once every 8 hours is adequate.

(continued)

BASES

ACTIONS (continued)

C.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits (Ref. 1). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions C.1.1 and C.1.2 or reduce power to ≤ 50% RTP.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that the RPI agrees with the demand position within [12] steps ensures that the RPI is operating correctly. Since the URPI does not display the actual shutdown rod positions between 18 and 210 steps, only points within the indicated ranges are required in comparison.

the requirements established in LCO 3.1.4

The [18 month] Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for unnecessary plant transients if the SR were performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed at a Frequency of once every [18 months.] Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

INSERT B3.1.7-2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13 UFSAR Section 3.1.2
2. FSAR, Chapter [15] CP&L Letter, E.E. Utley to NRC, Rod Position Indication System, dated 12/14/79
3. UFSAR, Chapter 15.

ITS INSERT B3.1.7-2

The Frequency of once prior to criticality after each removal of the reactor vessel head is based on the need to perform this Surveillance following events that could cause disturbance of the instruments. Performing the SR under the conditions that apply prior to criticality prevents the potential for unnecessary plant transients that could occur if the SR was performed with the reactor at power.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.9 PHYSICS TESTS Exceptions – MODE 1

BASES

BACKGROUND

The primary purpose of the MODE 1 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow the performance of instrumentation calibration tests and special PHYSICS TESTS. The exceptions to LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)" are most often appropriate for xenon stability tests. The exceptions to LCO 3.1.5, "Rod Group Alignment Limits"; LCO 3.1.6, "Shutdown Bank Insertion Limit"; and LCO 3.1.7, "Control Bank Insertion Limits," may be required in the event that it is necessary or desirable to do special PHYSICS TESTS involving abnormal rod or bank configurations. 3

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment at the facility has been accomplished, in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power, power

(continued)

BASES

BACKGROUND (continued)

ascension, and at power operation; and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 4).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation.

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 1 are listed below:

a. Neutron Flux Symmetry;

b. Power Distribution – Intermediate Power;

c. Power Distribution – Full Power; and

d. Critical Boron Concentration – Full Power.

The first test can be performed in either MODE 1 or 2, and the last three tests are performed in MODE 1. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance. The last two tests are performed at ≥ 90% RTP.

[a. The Neutron Flux Symmetry Test measures the degree of azimuthal symmetry of the core neutron flux at as low a power level as practical, depending on the method used. The Flux Distribution Method uses incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% RTP.

b. The Power Distribution – Intermediate Power Test measures the power distribution of the reactor core at intermediate power levels between 40% and 75% RTP. This test uses the incore flux detectors to measure core power distribution.]

3

(continued)

BASES

BACKGROUND (continued)

c. The Power Distribution - Full Power Test measures the power distribution of the reactor core at ≥ 90% RTP using incore flux detectors.

d. The Critical Boron Concentration - Full Power Test simply measures the critical boron concentration at > 90% RTP, with all rods fully withdrawn, the lead control bank being at or near its fully withdrawn position, and with the core at equilibrium xenon conditions.

For initial startups, there are two currently required tests that violate the referenced LCO. The pseudo ejected rod test, performed at approximately 30% RTP, and the pseudo dropped rod test, performed at approximately 50% RTP, require individual rod misalignments that exceed the limits specified in the relevant LCO.

3

APPLICABLE SAFETY ANALYSES

The fuel is protected by an LCO, which preserves the initial conditions of the core assumed during the safety analyses. The methods for development of the LCO, which are superseded by this LCO, are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating controls or process variables to deviate from their LCO limitations.

Reference 6 defines requirements for initial testing of the facility, including PHYSICS TESTS. Tables [14.1-1 and 14.1-2] (Ref. 6) summarize the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in:

LCO 3.1.5, "Rod Group Alignment Limits";
LCO 3.1.6, "Shutdown Bank Insertion Limits";
LCO 3.1.7, "Control Bank Insertion Limits";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; or

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)"

are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the requirements of LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," and LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," are satisfied, power level is maintained ≤ 85% RTP, and SDM is ≥ [1.6]% Δk/k. Therefore, LCO 3.1.9 requires surveillance of the hot channel factors and SDM to verify that their limits are not being exceeded.

PHYSICS TESTS include measurements of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the component and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

Reference 7 allows special test exceptions to be included as part of the LCO that they affect. However, it was decided to retain this special test exception as a separate LCO because it was less cumbersome and provided additional clarity.

LCO

This LCO allows selected control rods and shutdown rods to be positioned outside their specified alignment limits and insertion limits to conduct PHYSICS TESTS in MODE 1, to verify certain core physics parameters. The power level is limited to ≤ 85% RTP and the power range neutron flux trip setpoint is set at 10% RTP above the PHYSICS TESTS power level with a maximum setting of 90% RTP. Violation of LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, LCO 3.2.3, or LCO 3.2.4, during the performance of PHYSICS TESTS does not pose any threat to the integrity of the fuel as long as the requirements of LCO 3.2.1 and LCO 3.2.2 are satisfied and provided:

(continued)

BASES

LCO (continued)

a. THERMAL POWER is maintained ≤ 85% RTP;

b. Power Range Neutron Flux – High trip setpoints are ≤ 10% RTP above the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP; and

c. SDM is ≥ [1.6]% Δk/k.

Operation with THERMAL POWER ≤ 85% RTP during PHYSICS TESTS provides an acceptable thermal margin when one or more of the applicable LCOs is out of specification. The Power Range Neutron Flux – High trip setpoint is reduced so that a similar margin exists between the steady state condition and the trip setpoint that exists during normal operation at RTP.

APPLICABILITY

This LCO is applicable in MODE 1 when performing PHYSICS TESTS. The applicable PHYSICS TESTS are performed at ≤ 85% RTP. Other PHYSICS TESTS are performed at full power but do not require violation of any existing LCO, and therefore do not require a PHYSICS TESTS exception. The PHYSICS TESTS performed in MODE 2 are covered by LCO 3.1.10, "PHYSICS TESTS Exceptions – MODE 2."

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

When THERMAL POWER is > 85% RTP, the only acceptable actions are to reduce THERMAL POWER to ≤ 85% RTP or to suspend the PHYSICS TESTS exceptions. With the PHYSICS TESTS exceptions suspended, the PHYSICS TESTS may proceed if all other LCO requirements are met. Fuel integrity may be challenged with control rods or shutdown rods misaligned and THERMAL POWER > 85% RTP. The allowed Completion Time of 1 hour is reasonable, based on operating experience, for completing the Required Actions in an orderly manner and without challenging plant systems. This Completion Time is also consistent with the Required Actions of the LCOs that are suspended by the PHYSICS TESTS.

C.1 and C.2

When the Power Range Neutron Flux - High trip setpoints are > 10% RTP above the PHYSICS TESTS power level or > 90% RTP, the Reactor Trip System (RTS) may not provide the required degree of core protection if the trip setpoint is greater than the specified value.

The only acceptable actions are to restore the trip setpoint to the allowed value or to suspend the performance of the PHYSICS TESTS exceptions. The Completion Time of 1 hour is based on the practical amount of time it may take to restore the Neutron Flux - High trip setpoints to the correct value, consistent with operating plant safety. This Completion Time is consistent with the Required Actions of the LCOs that are suspended by the PHYSICS TESTS.

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that the THERMAL POWER level is ≤ 85% RTP will ensure that the required core protection is provided during the performance of PHYSICS TESTS. Control of the reactor power level is a vital parameter and is closely monitored during the performance of PHYSICS TESTS. A Frequency of 1 hour is sufficient for ensuring that the power level does not exceed the limit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.9.2

Verification of the Power Range Neutron Flux – High trip setpoints within 8 hours prior to initiation of the PHYSICS TESTS will ensure that the RTS is properly set to perform PHYSICS TESTS.

3

SR 3.1.9.3

The performance of SR 3.2.1.1 and SR 3.2.2.1 measures the core $F_Q(Z)$ and the $F^N_{\Delta H}$, respectively. If the requirements of these LCOs are met, the core has adequate protection from exceeding its design limits, while other LCO requirements are suspended. The Frequency of 12 hours is based on operating experience and the practical amount of time that it may take to run an incore flux map and calculate the hot channel factors.

SR 3.1.9.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. Reactor Coolant System (RCS) boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in the calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS. The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident without the required SDM.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. Regulatory Guide 1.68, Revision 2, August 1978.

4. ANSI/ANS-19.6.1-1985, December 13, 1985.

5. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report", July 1985.

6. FSAR, Section [14].

7. WCAP-11618, November 1987, and Addendum 1, April 1989.

3

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.10 PHYSICS TESTS Exceptions - MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include

(continued)

BASES

BACKGROUND (continued)

all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

The PHYSICS TESTS required for reload fuel cycles ~~(Ref. 4)~~ in MODE 2 are listed below:

a. Critical Boron Concentration - Control Rods Withdrawn;

b. Critical Boron Concentration - Control Rods Inserted;

c. Control Rod Worth;

d. Isothermal Temperature Coefficient (ITC); and

~~e. Neutron Flux Symmetry.~~

~~The first four tests are performed in MODE 2, and the last test can be performed in either MODE 1 or 2.~~ These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

a. The Critical Boron Concentration - Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical (k_{eff} = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.

b. The Critical Boron Concentration - Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% $\Delta k/k$ when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron

(continued)

BASES

BACKGROUND (continued)

concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.5, "Rod Group Alignment Limits"; LCO 3.1.6, "Shutdown Bank Insertion Limit"; or LCO 3.1.7, "Control Bank Insertion Limits."

c. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.5, LCO 3.1.6, or LCO 3.1.7.

d. The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of

(continued)

BASES

BACKGROUND (continued)

performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

e. The Flux Symmetry Test measures the degree of azimuthal symmetry of the neutron flux at as low a power level as practical, depending on the test method employed. This test can be performed at HZP (Control Rod Worth Symmetry Method) or at ≤ 30% RTP (Flux Distribution Method). The Control Rod Worth Symmetry Method inserts a control bank, which can then be withdrawn to compensate for the insertion of a single control rod from a symmetric set. The symmetric rods of each set are then tested to evaluate the symmetry of the control rod worth and neutron flux (power distribution). A reactivity computer is used to measure the control rod worths. Performance of this test could violate LCO 3.1.5, LCO 3.1.6, or LCO 3.1.7. The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% RTP.

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

problems, may require the operating control or process variables to deviate from their LCO limitations.

The FSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. Tables 14.1-1 and 14.1-2 summarize the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.4, "Moderator Temperature Coefficient (MTC)," LCO 3.1.5, LCO 3.1.6, LCO 3.1.8, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is kept ≥ 531°F, and SDM is ≥ 1.6% Δk/k.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR. PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

Reference 6 allows special test exceptions (STEs) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified

(continued)

BASES

LCO (continued) limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 531 °F; and

b. SDM is ≥ 1.6% Δk/k. within the limits provided in the COLR; and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY This LCO is applicable in MODE 2 when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP. Other PHYSICS TESTS are performed in MODE 1 and are addressed in LCO 3.1.9, "PHYSICS TESTS Exceptions – MODE 1."

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

(continued)

BASES

ACTIONS (continued)

C.1

When the RCS lowest T_{avg} is < 531°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 531°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.10.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel within 12 hours prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The 12 hour time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

SR 3.1.10.2

Verification that the RCS lowest loop T_{avg} is ≥ 531°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.10.2 (continued)

performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.10.3

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The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. Regulatory Guide 1.68, Revision 2, August, 1978.

4. ANSI/ANS-19.6.1-1985, December 13, 1985.

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(continued)

ITS INSERT B3.1.8-1

SR 3.1.8.2

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of THERMAL POWER at a frequency of 30 minutes during the performance of the PHYSICS TEST will ensure that the initial conditions of the safety analyses are not violated.

BASES

REFERENCES (continued)

5. ~~WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.~~

6. ~~WCAP-11618, including Addendum 1, April 1989.~~

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.11 SHUTDOWN MARGIN (SDM) Test Exception

BASES

BACKGROUND

The primary purpose of the SDM test exception is to permit relaxation of the SDM requirements during the measurement of control rod worths in MODE 2 during PHYSICS TESTS.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment at the facility has been accomplished in accordance with the design; and

e. Verify that operating and emergency procedures are adequate.

To achieve these objectives, testing is performed prior to initial criticality, during startup, low power, power ascension, and at power operation, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

PHYSICS TEST procedures are written and approved, in accordance with established formats. The procedures include

(continued)

BASES

BACKGROUND (continued)

all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation.

During the PHYSICS TESTS measurements of control rod worth, it may be necessary to align individual rods and banks in certain configurations and utilize boron concentrations that do not provide sufficient SDM to meet the normal requirements. In this situation, it is necessary to invoke special test exceptions (STEs) to allow the necessary PHYSICS TESTS to be completed.

APPLICABLE SAFETY ANALYSES

Special PHYSICS TESTS may require operating the core under controlled conditions for short periods of time with less than the normally required SDM. As such, these tests are not covered by any safety analysis calculations.

Under the acceptance criteria to allow suspension of certain LCOs for PHYSICS TESTS, fuel damage criteria are not to be exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because adequate limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 5 defines the requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). PHYSICS TESTS for reload fuel cycles are given in Table 1 of ANSI/ANS-19-6.1-1985. Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, Conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the linear heat rate remains within its limit, fuel design criteria are preserved.

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

3

(continued)

BASES (continued)

LCO

This LCO provides an exemption to the SDM requirements under controlled conditions. These conditions require that at least the highest estimated control rod worths be available for trip insertion. It is assumed that this available negative reactivity will be sufficient to shut down the core if required, assuming there is not a concurrent boron dilution or cooldown event. This exemption is allowed even though there are no bounding safety analyses, because the tests are performed under close supervision and provide valuable information on control rod worth and core SDM. 3

APPLICABILITY

This LCO is only applicable in MODE 2, and then only during actual measurement of control rod worths because this is the only time the exception is required.

ACTIONS

A.1

If one or more control rods are not fully inserted and the available trip reactivity from OPERABLE control rods is less than the highest estimated control rod worth, the SDM, assumed for the test conditions, may not be available. Under these conditions, it is necessary to promptly restore the SDM to within limits.

The allowed Completion Time of 15 minutes ensures prompt action and provides an acceptable time for initiating boration to restore SDM, without allowing the core to remain in an unacceptable condition for an extended period of time.

B.1

If all control rods are fully inserted, and the reactor is subcritical by less than the highest estimated control rod worth, the SDM, assumed for the test conditions, may not be available. Under these conditions, it is necessary to promptly restore the SDM to within limits.

The allowed Completion Time of 15 minutes provides an acceptable time for initiating boration to restore SDM, without allowing the core to remain in an unacceptable condition for an extended period of time.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.11.1

In order to establish an acceptable SDM during the measurement of control rod worths, it is necessary to know the position of each control rod. A test Frequency of 2 hours is reasonable, based on normal control rod motion during control rod worth measurements.

SR 3.1.11.1 has been modified by a Note establishing that the position of only those control rods not fully inserted must be determined. It is assumed that the position and worth of fully inserted control rods is known.

3

SR 3.1.11.2

One of the assumptions made in granting an STE for SDM, is that all control rods not fully inserted will fully insert when tripped. This Surveillance is performed to verify that fact.

The Frequency of 24 hours prior to reducing the plant SDM below the normal requirements is acceptable, based on the assumption that the control rods will remain OPERABLE and trippable for 24 hours and during the performance of the test.

SR 3.1.11.2 has been modified by a Note establishing that this Surveillance is only required for control rods not fully inserted. During the performance of control rod worth measurements, certain control rods remain fully inserted. Since these rods are not relied on to trip, there is no need to demonstrate that they will fully insert when tripped.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. Regulatory Guide 1.68, Revision 2, August 1978.
4. ANSI/ANS-19.6.1-1985, December 13, 1985.
5. FSAR, Chapter [14].

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.1 - REACTIVITY CONTROL SYSTEMS

PART 7

JUSTIFICATION FOR DIFFERENCES (JFDs) TO ISTS BASES

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes which involve the insertion of plant specific terms or parameters are used to preserve consistency with the CTS and licensing basis.

2 HBRSEP was designed and licensed to the proposed Appendix A to 10 CFR 50, which was published in the Federal Register on July 11, 1967 (FR 32FR10213). Appendix A to 10 CFR 50 effective in 1971 and subsequently amended, is somewhat different from the proposed 1967 criteria. UFSAR section 3.1 includes an evaluation of HBRSEP with respect to the proposed 1967 criteria. The ISTS statement concerning the GDC criteria is modified in the ITS to reference the current licensing basis description in the UFSAR.

3 ISTS Specification 3.1.2 is not included as a separate specification in the ITS. Since the specific shutdown margin requirements for various plant conditions are relocated to the Core Operating Limits Report (COLR), there is no need for separate specifications for different MODES of Applicability. Consequently, shutdown margin requirements applicable to MODE 5 are included in ITS Specification 3.1.1. This eliminates the need for Specification 3.1.2. ISTS Specifications 3.1.9 and 3.1.11 are also not adopted in the ITS. Subsequent Specifications are renumbered accordingly.

4 The phrase, "... and the fuel and moderator temperatures are changed to the nominal hot zero power value ...," is added to clarify the assumptions used in determining the shutdown margin requirements during operation.

5 The term, "Control Rod System," is replaced with the phrase, "two independent reactivity control systems," to clarify that power maneuvers require both the control rod system and the Chemical and Volume Control System in concert (i.e., for boron concentration changes) to maintain the core flux shape within the axial and radial differential limitations.

6 The terms, "soluble boron system," and "boration system," are replaced with the plant specific terminology, "Chemical and Volume Control System," or "CVCS."

7 The phrase, "... Rod Cluster Control Assemblies and ...," is added because the worth of the control rod banks provide an essential portion of the shutdown margin.

8 The inserted phrase is relocated from ISTS page B 3.1-8, to address SDM in MODE 5.

9 The phrase, "... until the MODE 5 value is reached ...," is deleted because, in some cases (e.g., main steam line break inside containment) the containment will pressurize, resulting in a saturation temperature of >200°F for the faulted steam generator (the MODE 5 value).

10 The safety analysis indicates that postulated control rod withdrawal events are terminated by either a high flux or overtemperature ΔT trip.

11 The phrase, "... and operator response time ...," is added, because the UFSAR Chapter 15 safety analysis assumes that manual action is taken by the operator.

12 Plant specific terminology, "refueling water storage tank," is inserted in place of "borated water storage tank."

13 The phrase, "... may approach or exceed 2000 ppm ...," is replaced by, "... is greatest ...," in order to better relay the intent of the Bases statement. Because the boron concentration is greatest at the beginning of core life, a given volume of concentrated boric acid solution added to the RCS will have the least impact on boron concentration and reactivity at this time. The values of boric acid flow, boric acid concentration, and time to increase boron concentration 100 ppm, for the example given, are changed to those that reflect plant specifics.

14 Bases are modified by adding the phrases, "... with keff $\geq$1.0 ..." and "... MODE 2 with K_{eff} < 1.0 and ...," to agree with the Applicability for LCO 3.1.1.

15 The method used to determine shutdown margin involves comparison of the RCS boron concentration to the SDM curve. The SDM curve defines the boron concentration necessary to provide the one percent SDM for the fuel reactivity indicated by the previous critical boron concentration.

16 The term, "critical boron curve," is not used at HBRSEP.

17 Bases are modified by adding the term, "or equal to," to agree with the limits specified in the COLR.

18 The UFSAR Section 15.1.5 safety analysis indicates a return to power is possible after a main steam line break.

19 With increased fuel enrichment, the beginning of core life MTC has become more positive than the original accident analysis contained in the Final Facility Description and Safety Analysis Report. This has resulted in reanalysis of some UFSAR accidents. Reference to original accident analysis is eliminated, since it is no longer bounding.

20 The phrase, "assumed in the most limiting accident analysis," is added to clarify the acceptance criteria for the 300 ppm MTC.

21 The term, "more positive," is replaced with the term, "less negative," to be consistent with the wording of the SR.

22 WCAP 9273-NP-A, July 1985, is not a valid reference.

23 Control rods are normally fully withdrawn during power operations.

24 A redundant rod position indication system is not part of the plant design.

25 The static rod misalignment analysis includes insertion of a Bank beyond its insertion limit.

26 Rod misalignment limits consist of two values, which are dependent on the bank demand position. These alignment limits are consistent with the current licensing basis. These values are specified in LCO 3.1.4, and need not be repeated in the Bases.

27 The term, "bottomed," is replaced with the term, "inserted." The rods are not necessarily fully inserted (i.e., bottomed) when in MODES 3, 4, 5, and 6.

28 Bases for ITS Specification 3.1.4, Required Action B.2.2, which requires reducing THERMAL POWER to ≤ 75 percent RTP, is modified to specify reducing THERMAL POWER to ≤70% RTP, consistent with current licensing basis.

29 The term, "obviates," is changed to the more commonly used term, "eliminates."

30 The Bases for SR 3.1.4.3 is modified to reflect a minimum T_{avg} of 540°F for verification of rod drop times, consistent with current licensing basis.

31 The Bases for Specification 3.1.5 are modified to reflect that the positions of the control banks are not necessarily normally controlled by the Rod Control System.

32 Control rods are not fully inserted during cooldown. All rods are withdrawn to five steps to allow for thermal contraction and to prevent rods from jamming inside the dashpots.

33 Plant specific values are inserted in the Bases 3.1.6 discussion. A different example Figure B 3.1.6-1 is inserted to reflect that maximum rod withdrawal is 225 steps, and not 231 steps.

34 The OPERABILITY requirements are specified in LCO 3.1.7, and need not be repeated in the Bases.

35 Demand position indication is not calibrated. The counters are reset to zero when rods are fully inserted prior to startup.

36 The referenced analysis does not include explicit consideration of the effects on core peaking factors of rod position versus power level, and is not retained in the ITS.

37 Bases for Specification 3.1.7, Required Action C.1.2, is modified to be consistent with ITS Specification 3.1.4, as modified, and current licensing basis.

38 Shutdown bank rod position indication is provided over the full range.

39 HBRSEP is not committed to either Regulatory Guide 1.68 or ANSI/ANS-19.6.1.

40 The word, "more," is changed to the word, "both," because plant design includes two shutdown banks.

41 The boron exchange methodology is the method used at HBRSEP to perform integral and differential rod worth measurements. This method is used to determine the reactivity of individual rod banks, as well as the reactivity of the predicted "worst case" stuck rod.

42 The "average slope method" is used at HBRSEP for measuring isothermal temperature coefficient (ITC).

43 ITS SR 3.1.8.1 is deleted. Performance of a COT on power range and intermediate range channels is required by LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," every 92 days (SR 3.3.1.7 and SR 3.3.1.8). The 92 day required Frequency has been determined to be sufficient for verification that the power range and intermediate range monitors are properly functioning.

44 The referenced reports are not applicable to HBRSEP.

45 Not used.

46 Not used.

47 Bases are modified for consistency with the scope and content of the associated Specification. This change is based on the need to perform the surveillance following plant evolutions that could cause disturbance of the instruments.

48 Bases 3.1.4 for ACTIONS is modified to correct error due to incomplete incorporation of a Revision 0 change (WOG-17, C.1). The Bases now match the Required Actions.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.1 - REACTIVITY CONTROL SYSTEMS

PART 8

PROPOSED HBRSEP, UNIT NO. 2 ITS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limits provided in the COLR.

APPLICABILITY: MODE 2 with $k_{eff} < 1.0$,
MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM is within the limits provided in the COLR	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

LCO 3.1.2 The measured core reactivity shall be within ± 1% Δk/k of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Measured core reactivity not within limit.	A.1	Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	72 hours
	AND		
	A.2	Establish appropriate operating restrictions and SRs.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	----------NOTE---------- The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. ---------- Verify measured core reactivity is within ± 1% Δk/k of predicted values.	Once prior to entering MODE 1 after each refueling AND -----NOTE------ Only required after 60 EFPD ---------- 31 EFPD thereafter

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR. The maximum upper limit shall be ≤ +5.0 pcm/°F at less than 50% RTP or 0.0 pcm/°F at 50% RTP and above.

APPLICABILITY: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ for the upper MTC limit, MODES 1, 2, and 3 for the lower MTC limit.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. MTC not within upper limit.	A.1	Establish administrative withdrawal limits for control banks to maintain MTC within limit.	24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 2 with $k_{eff} < 1.0$.	6 hours
C. MTC not within lower limit.	C.1	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify MTC is within upper limit.	Once prior to entering MODE 1 after each refueling
SR 3.1.3.2	----------NOTES---------- 1. Not required to be performed until 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm. 2. If the MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR, SR 3.1.3.2 shall be repeated once per 14 EFPD during the remainder of the fuel cycle. 3. SR 3.1.3.2 need not be repeated if the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR. ---------- Verify MTC is within lower limit.	Once each cycle

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE.

AND

Individual indicated rod positions shall be as follows:

a. For bank demand positions ≥ 200 steps, each rod shall be within 15 inches of its bank demand position, and

b. For bank demand positions < 200 steps, each rod shall be within 7.5 inches of the average of the individual rod positions in the bank.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) inoperable.	A.1.1 Verify SDM is within the limits provided in the COLR.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Be in MODE 3.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One rod not within alignment limits.	B.1 Restore rod to within alignment limits.	1 hour
	OR	
	B.2.1.1 Verify SDM is within the limits provided in the COLR.	1 hour
	OR	
	B.2.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	B.2.2 Reduce THERMAL POWER to ≤ 70% RTP.	2 hours
	AND	
	B.2.3 Verify SDM is within the limits provided in the COLR.	Once per 12 hours
	AND	
	B.2.4 Perform SR 3.2.1.1.	72 hours
	AND	
	B.2.5 Perform SR 3.2.2.1.	72 hours
	AND	
	B.2.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is within the limits provided in the COLR.	1 hour
	OR	
	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	AND	
	D.2 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 1.8 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 540°F; and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limits

LCO 3.1.5 Each shutdown bank shall be within insertion limits specified in the COLR.

APPLICABILITY: MODE 1,
MODE 2 with any control bank not fully inserted.

----------NOTE----------
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both shutdown banks not within limits.	A.1.1 Verify SDM is within the limits provided in the COLR.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Restore shutdown banks to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each shutdown bank is within the limits specified in the COLR.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

LCO 3.1.6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

----------NOTE----------
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Control bank insertion limits not met.	A.1.1 Verify SDM is within the limits provided in the COLR.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Restore control bank(s) to within limits.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Control bank sequence or overlap limits not met.	B.1.1 Verify SDM is within the limits provided in the COLR.	1 hour
	OR	
	B.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	B.2 Restore control bank sequence and overlap to within limits.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify estimated critical control bank position is within the limits specified in the COLR.	Within 4 hours prior to achieving criticality

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.6.2	Verify each control bank insertion is within the limits specified in the COLR.	12 hours
SR 3.1.6.3	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Analog Rod Position Indication (ARPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each inoperable rod position indicator per group and each demand position indicator per bank.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One ARPI per group inoperable for one or more groups.	A.1	Verify the position of the rods with inoperable position indicators by using movable incore detectors.	Once per 8 hours
	OR		
	A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. One or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	B.1	Verify the position of the rods with inoperable position indicators by using movable incore detectors.	6 hours
	OR		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
C. One demand position indicator per bank inoperable for one or more banks.	C.1.1	Verify by administrative means all ARPIs for the affected banks are OPERABLE.	Once per 8 hours
	AND		
	C.1.2	Verify the position of each rod in the affected bank(s) is within 7.5 inches of the average of the individual rod positions in the affected bank(s).	Once per 8 hours
	OR		
	C.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify each ARPI agrees within the requirements established in LCO 3.1.4 of the group demand position for the full indicated range of rod travel.	Once prior to criticality after each removal of the reactor vessel head.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.4, "Rod Group Alignment Limits";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended, provided:

a. RCS lowest loop average temperature is ≥ 530°F;

b. SDM is within the limits provided in the COLR; and,

c. THERMAL POWER is ≤ 5% RTP

APPLICABILITY: During PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. SDM not within limit.	A.1	Initiate boration to restore SDM to within limit.	15 minutes
	AND		
	A.2	Suspend PHYSICS TESTS exceptions.	1 hour
B. THERMAL POWER not within limit.	B.1	Open reactor trip breakers.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Verify the RCS lowest loop average temperature is ≥ 530°F.	30 minutes
SR 3.1.8.2	Verify THERMAL POWER is ≤ 5% RTP.	30 minutes
SR 3.1.8.3	Verify SDM is within the limits provided in the COLR.	24 hours

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.1 - REACTIVITY CONTROL SYSTEMS

PART 9

PROPOSED BASES TO HBRSEP, UNIT NO. 2 ITS

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to HBRSEP Design Criteria (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single control rod assembly of highest reactivity worth is fully withdrawn and the fuel and moderator temperatures are changed to the normal hot zero power value.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control rod assemblies and soluble boric acid in the Reactor Coolant System (RCS). The two independent reactivity control systems can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Control Rod System, together with the Chemical and Volume Control System (CVCS), provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The CVCS can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and

(continued)

BASES

BACKGROUND (continued)

refueling modes, the SDM requirements are met by means of rod cluster control assemblies and adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in the safety analyses. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out following a reactor scram. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that the specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 280 cal/gm energy deposition for the rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition;

c. Startup of an inactive reactor coolant pump (RCP); and

d. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS, the assumed dilution flow rate and operator response time, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high flux level trip or an overtemperature ΔT trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the minimum required time (Ref. 5) assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 2 with $k_{eff} < 1.0$ and in MODES 3 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6.

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that

(continued)

BASES

ACTIONS

A.1 (continued)

boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tanks, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration is greatest. Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of 60 gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 5 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% $\Delta k/k$. These boration parameters of 60 gpm and 21000 ppm represent typical values and are provided for the purpose of offering a specific example (Ref. 6).

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2 with $K_{eff} \geq 1.0$, SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

In MODE 2 with $K_{eff} < 1.0$ and MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance verification, considering the listed reactivity effects:

a. RCS boron concentration;

b. Control bank position;

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

c. RCS average temperature;

d. Fuel burnup based on previous critical boron concentration;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the verification.

REFERENCES

1. UFSAR, Section 3.1.

2. UFSAR, Section 15.1.5.

3. UFSAR, Section 15.4.6.

4. 10 CFR 100.

5. UFSAR, Table 15.4.6-1.

6. UFSAR, Table 9.3.4-1.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Core Reactivity

BASES

BACKGROUND

According to HBRSEP Design Criteria (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during critical operationS. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve, which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the

(continued)

BASES

BACKGROUND (continued)

calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of the NRC Policy Statement.

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of ± 1% $\Delta k/k$ has been established based on engineering judgement. A 1% deviation in reactivity from

(continued)

BASES

LCO (continued)

that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1% Δk/k of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. An SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 72 hours is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then the boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by a Note. The Note indicates that the normalization of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly.

REFERENCES

1. UFSAR Section 3.1.

2. UFSAR, Chapter 15.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to HBRSEP Design Criteria (Ref. 1), the reactor core with its related controls and protection systems are designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than or equal to zero when THERMAL POWER is at RTP. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

(continued)

BASES

BACKGROUND (continued)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the UFSAR accident and transient analyses.

If the LCO limits are not met, the unit response during transients may not be as predicted. The core design could violate the departure from nucleate boiling ratio criteria of the approved correlation, which could lead to a loss of the fuel cladding integrity.

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The UFSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding.

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from either zero (Ref. 3) or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the BOC or EOC life. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. An EOC measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

MTC satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.3 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at BOC; this upper bound must not be exceeded. This maximum upper limit occurs at BOC, all rods out (ARO), hot zero power conditions. At EOC the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

(continued)

BASES

LCO (continued)

The LCO establishes a maximum positive value that cannot be exceeded. The BOC positive limit and the EOC negative limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.

APPLICABILITY

Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled CONTROL ROD assembly or group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

If the BOC MTC limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits.

(continued)

BASES

ACTIONS

A.1 (continued)

As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

B.1

If the required administrative withdrawal limits at BOC are not established within 24 hours, the unit must be brought to MODE 2 with $k_{eff} < 1.0$ to prevent operation with an MTC that is more positive than that assumed in safety analyses.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

C.1

Exceeding the EOC MTC limit means that the safety analysis assumptions for the EOC accidents that use a bounding negative MTC value may be invalid. If the EOC MTC limit is exceeded, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 4 within 12 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

This SR requires measurement of the MTC at BOC prior to entering MODE 1 in order to demonstrate compliance with the most positive MTC LCO. Meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOC MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the BOC MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

SR 3.1.3.2

In similar fashion, the LCO demands that the MTC be less negative than the specified value assumed in the most limiting accident analysis for EOC full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the EOC LCO limit. The 300 ppm SR value is sufficiently less negative than the EOC LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met.

SR 3.1.3.2 is modified by three Notes that include the following requirements:

a. The SR is not required to be performed until 7 effective full power days (EFFDs) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.2 (continued)

b. If the 300 ppm Surveillance limit is exceeded, it is possible that the EOC limit on MTC could be reached before the planned EOC. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the EOC limit.

c. The Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is less negative than the 60 ppm Surveillance limit, the EOC limit will not be exceeded because of the gradual manner in which MTC changes with core burnup.

REFERENCES

1. UFSAR Section 3.1.

2. UFSAR, Section 15.0.5.

3. UFSAR, Section 15.4.1.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are described in the UFSAR (Ref. 1) and 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups

(continued)

BASES

BACKGROUND (continued)

that are moved in a staggered fashion, but always within one step of each other. HBRSEP has four control banks and two shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues for the remaining control banks. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Analog Rod Position Indication (ARPI) System.

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The ARPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is six steps.

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BASES

BACKGROUND (continued)

The maximum uncertainty of the ARPI System is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and ARPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches (Ref. 4 and 6).

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

There be no violations of:

a. specified acceptable fuel design limits, or
b. Reactor Coolant System (RCS) pressure boundary integrity.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Two types of analysis are performed in regard to static rod misalignment (Ref. 3). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted in excess of its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 5).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat generation rates (LHGRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of the NRC Policy Statement.

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e., trippability to meet SDM) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required time on a valid signal. CRDM malfunctions that result in inability to move a rod (e.g., rod urgent failures), which do not impact trippability, do not necessarily result in rod inoperability.

The requirement to maintain the rod alignment to within the specified limits is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in

(continued)

BASES

LCO (continued)

some cases a total misalignment from fully withdrawn to fully inserted is assumed.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHGRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are inserted and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

ACTIONS

A.1.1 and A.1.2

When one or more rods are inoperable (e.g., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM.

In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

(continued)

BASES

ACTIONS (continued)

A.2

If the inoperable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1

When a rod becomes misaligned, it can usually be moved and is still trippable. If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be moved fully in and control bank C must be moved in to approximately 100 to 115 steps.

(continued)

BASES

ACTIONS

B.2.2.1 and B.2.1.2 (continued)

Power operation may continue with one RCCA trippable but misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 70% RTP ensures that local LHGR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 7). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Completion Time of once per 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 70% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

(continued)

BASES

ACTIONS (continued)

C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which eliminates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

(continued)

BASES

ACTIONS

D.2 (continued)

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. If the rod position deviation monitor in inoperable, a Frequency of 4 hours accomplishes the same goal. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

SR 3.1.4.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable by the normal CRDM, but remains trippable, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable by the normal CRDM, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 540°F to simulate a reactor trip under actual conditions.

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. UFSAR Section 3.1.
2. 10 CFR 50.46.
3. UFSAR Section 15.4.
4. CP&L Letter, E.E. Utley to NRC, Rod Position Indication System, dated December 14, 1979.
5. UFSAR, Section 15.0.6.
6. NRC Letter to CP&L, Mr. J. A. Jones, "Amendment No. 48 to Facility Operating License No. DPR-23 for HBRSEP, Unit No. 2," dated August 29, 1979.
7. UFSAR, Section 15.4.3.2.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are described in the UFSAR (Ref. 1) and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. HBRSEP has four control banks and two shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks can be automatically controlled by the Rod Control System, or they can be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations.

(continued)

BASES

BACKGROUND (continued)

Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown banks are then left in this position until the reactor is shut down. They affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

There be no violations of:

a. specified acceptable fuel design limits, or
b. RCS pressure boundary integrity.

As such, the shutdown bank insertion limits affect safety analysis involving core reactivity and SDM (Ref. 3).

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of the NRC Policy Statement.

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

APPLICABILITY

The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. The applicability in MODE 2 begins prior to initial control bank withdrawal, during an approach to criticality, and continues throughout MODE 2, until all control bank rods are again fully inserted by reactor trip or by shutdown. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6, the shutdown banks contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.2. This SR verifies the freedom of the rods to

(continued)

BASES

APPLICABILITY (continued)

move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1.1, A.1.2 and A.2

When one or both shutdown banks is not within insertion limits, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1 (continued)

shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

REFERENCES

1. UFSAR, Section 3.1.

2. 10 CFR 50.46.

3. UFSAR, Chapter 15.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are described in the UFSAR (Ref. 1) and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. HBRSEP has four control banks and two shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. An example is provided for information only in Figure B 3.1.6-1. The control banks are required to be at or above the insertion limit lines.

Figure B 3.1.6-1 also indicates how the control banks are moved in an overlap pattern. Overlap is the distance travelled together by two control banks. The predetermined

(continued)

BASES

BACKGROUND (continued)

position of control bank C, at which control bank D will begin to move with bank C on a withdrawal, will be at 128 steps for a fully withdrawn position of 225 steps. The fully withdrawn position is defined in the COLR.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks can be controlled automatically by the Rod Control System, or they can be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.4, LCO 3.1.5, "Shutdown Bank Insertion Limits," LCO 3.1.6, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RPS trip function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

There be no violations of:

a. specified acceptable fuel design limits, or
b. Reactor Coolant System pressure boundary integrity.

As such, the shutdown and control bank insertion limits affect safety analysis involving core reactivity and power distributions (Ref. 3).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 3).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 3).

The insertion limits satisfy Criterion 2 of the NRC Policy Statement, in that they are initial conditions assumed in the safety analysis.

LCO

The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate

(continued)

BASES

LCO (continued)

negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

The applicability requirements have been modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO.

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2

When the control banks are outside the acceptable insertion limits, they must be restored to within those limits. This restoration can occur in two ways:

a. Reducing power to be consistent with rod position; or

b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

(continued)

BASES

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2 (continued)

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlaps limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

C.1

If the Required Actions cannot be completed within the associated Completion Times, the plant must be brought to MODE 3, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1 (continued)

that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.6.2

With an OPERABLE bank insertion limit monitor, verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY of the bank insertion limit monitor and to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours. If the insertion limit monitor becomes inoperable, verification of the control bank position at a Frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits.

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.

REFERENCES

1. UFSAR Section 3.1.

2. 10 CFR 50.46.

3. UFSAR, Chapter 15.

BASES

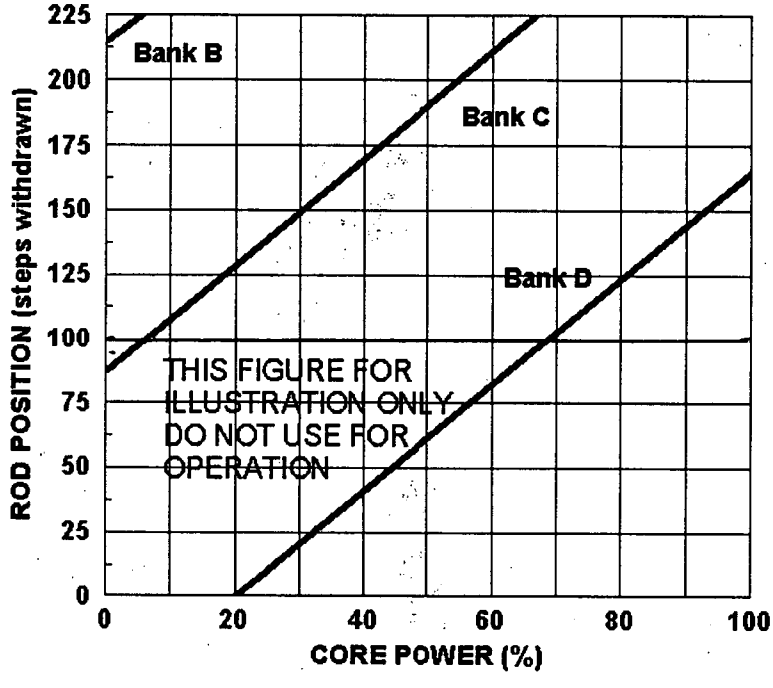

Figure B 3.1.6-1 (page 1 of 1)
Control Bank Insertion vs. Percent RTP

B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

The applicable design criteria for rod position indication described in the UFSAR (Ref. 1). LCO 3.1.7 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control.

(continued)

BASES

BACKGROUND (continued)

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the Analog Rod Position Indication (ARPI) System.

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± " inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The ARPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is 6 steps. Therefore, the normal indication accuracy of the ARPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and ARPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches (Ref. 2).

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 3), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits," and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of the NRC Policy Statement. The control rod position indicators monitor control rod position, which is an initial condition of the accident.

LCO

LCO 3.1.7 specifies that one ARPI System and one Bank Demand Position Indication System be OPERABLE for each control rod. For the control rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following:

a. The ARPI System meets the requirements of LCO 3.1.4, "Rod Group Alignment Limits";

b. For the ARPI System there are no known failed coils; and

c. The Bank Demand Indication System had been previously reset to zero with all rods in the fully inserted position.

By meeting the requirements of LCO 3.1.4, the Bank Demand Position Indication System can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.4, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

(continued)

BASES

LCO (continued)

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements on the ARPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator per group and each demand position indicator per bank. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1

When one ARPI channel per group fails, the position of the rod can still be determined by use of the incore movable detectors. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of B.1 or B.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

(continued)

BASES

ACTIONS (continued)

A.2

Reduction of THERMAL POWER to ≤ 50% RTP more than offsets the increase in core F_Q and $F^N_{\Delta H}$ due to rod position.

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1 and B.2

These Required Actions clarify that when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 are still appropriate but must be initiated promptly under Required Action B.1 to begin verifying that these rods are still properly positioned, relative to their group positions.

If, within 6 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 6 hours provides an acceptable period of time to verify the rod positions.

C.1.1 and C.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the ARPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and that the position of each rod in the affected bank(s) is within 7.5 inches of the average of the individual rod positions in the affected bank(s), within the allowed Completion Time of once every 8 hours is adequate.

(continued)

BASES

ACTIONS (continued)

C.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors. The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions C.1.1 and C.1.2 or reduce power to ≤ 50% RTP.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the ARPI agrees with the demand position within the requirements established in LCO 3.1.4 ensures that the DRPI is operating correctly.

The Frequency of once prior to criticality after each removal of the reactor vessel head is based on the need to perform this Surveillance following events that could cause disturbance of the instruments. Performing the SR under the conditions that apply prior to criticality prevents the potential for unnecessary plant transients that could occur if the SR was performed with the reactor at power.

REFERENCES

1. UFSAR Section 3.1.2.

2. CP&L Letter, E. E. Utley to NRC, "Rod Position Indication System," dated 12/14/79.

3. UFSAR, Chapter 15.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions – MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to:

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed.

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include

(continued)

BASES

BACKGROUND (continued)

all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

The PHYSICS TESTS required for reload fuel cycles in MODE 2 are listed below:

a. Critical Boron Concentration – Control Rods Withdrawn;

b. Critical Boron Concentration – Control Rods Inserted;

c. Control Rod Worth;

d. Isothermal Temperature Coefficient (ITC); and

These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

a. The Critical Boron Concentration – Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical (k_{eff} = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.

b. The Critical Boron Concentration – Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% $\Delta k/k$ when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron

(continued)

BASES

BACKGROUND (continued)

concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.4, "Rod Group Alignment Limits"; LCO 3.1.5, "Shutdown Bank Insertion Limit"; or LCO 3.1.6, "Control Bank Insertion Limits."

c. The Control Rod Worth Test is used to measure the differential and integral reactivity worths of selected control banks or individual rods. This test is performed at HZP. The Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This data is used to determine the integral and differential worths of individual banks and rods. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6.

d. The ITC Test measures the ITC of the reactor. This test is performed at HZP and consists of varying RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

The UFSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. Table 14.2.6-2 summarizes the zero, low power, and power tests. Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.3, "Moderator Temperature Coefficient (MTC)," LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is kept ≥ 530°F, and SDM is within the limits provided in the COLR.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR. PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified

(continued)

BASES

LCO (continued)

limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 530 °F;

b. SDM is within the limits provided in the COLR; and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY

This LCO is applicable when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP.

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

(continued)

BASES

ACTIONS (continued)

C.1

When the RCS lowest T_{avg} is < 530°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 530°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that the RCS lowest loop T_{avg} is ≥ 530°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.2

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of THERMAL POWER at a frequency of 30 minutes during the performance of the PHYSICS TEST will ensure that the initial conditions of the safety analyses are not violated.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.3

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.1 - REACTIVITY CONTROL SYSTEMS

PART 10

ISTS GENERIC CHANGES

Industry/TSTF Standard Technical Specification Change Traveler

Move SR for 300 ppm MTC measurement to Frequency Note of SR 3.1.4.3

NUREGs Affected: ☐ 1430 ☒ 1431 ☐ 1432 ☐ 1433 ☐ 1434

Description:

Delete SR 3.1.4.2 (MTC measurement at 300 ppm) and replace it with a note in SR 3.1.4.3. Reformat the Bases appropriately.

Justification:

SR 3.1.4.2 was deleted since the intent of this SR is only to determine the next frequency for SR 3.1.4.2. Identifying this frequency determination as an SR implies that it must be met in order to meet the LCO per SR 3.0.1, which is not true for this limit. SR 3.1.4.3 currently contains a Note that addresses the accelerated frequency required if the 300 ppm Surveillance limit is not met. The Note in the Frequency column of SR 3.1.4.3 is moved to the Surveillance column of SR 3.1.4.3 for clarification purposes. This change would make the use of this type of Note consistent with other SRs and the examples contained in Section 1.4.

Affected Technical Specifications

SR 3.1.4.2	Moderator Temperature Coefficient (MTC)
SR 3.1.4.2 Bases	Moderator Temperature Coefficient (MTC)
SR 3.1.4.3	Moderator Temperature Coefficient (MTC)
SR 3.1.4.3 Bases	Moderator Temperature Coefficient (MTC)

WOG Review Information

WOG-4.5

Originating Plant: Date Provided to OG: 15-Mar-95 Needed By:

Owners Group History:

WOG-4, C.5

Owners Group Resolution: Approved Date: 11-Aug-95

TSTF Review Information

TSTF Received Date: 05-Sep-95 Date Distributed to OGs for Review: 05-Sep-95

OG Review Completed: ☒ BWOG ☒ WOG ☒ CEOG ☒ BWROG

TSTF History:

TSTF Resolution: Approved Date: 05-Sep-95 **TSTF- 13**

NRC Review Information

NRC Received Date: 03-Oct-95 NRC Reviewer: R. Tjader Reviewer Phone #:

Reviewer Comments:

10/4/95 - R. Tjader referred pkg to Tech Branch for review.
10/31/95 - change approved.
11/14/95 - pkg to TSB mgmt.
11/17/95 - TSB mgmt approved change.

Final Resolution: Approved Date: 27-Nov-95

Revision History

Revision 1 Revision Date: 08-Jan-96 Proposed by: TSTF

Revision Description:
Remarked the pages to use the TSTF number instead of the OG number.

Resolution: Date:

Incorporation Into the NUREGs

File to BBS/LAN Date:

File to TSTF Date:

File Rev Incoporated:

File Rev Incorporated Date

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify MTC is within upper limit.	Once prior to entering MODE 1 after each refueling
~~SR 3.1.4.2~~	~~Verify MTC is within 300 ppm Surveillance limit specified in the COLR.~~	~~NOTE~~ 1. Not required to be performed until 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm ~~Once each cycle~~
SR 3.1.4.~~2~~ 3	NOTES ~~1.~~ If the MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR, SR 3.1.4.3 shall be repeated once per 14 EFPD during the remainder of the fuel cycle. ~~2.~~ SR 3.1.4.~~3~~ 2 need not be repeated if the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR. Verify MTC is within lower limit.	~~NOTE~~ ~~Not required to be performed until 7 EFPD after reaching the equivalent of an equilibrium RTP-ARO boron concentration of 300 ppm~~ Once each cycle

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2 ~~and SR 3.1.4.3~~

In similar fashion, the LCO demands that the MTC be less negative than the specified value for EOC full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the EOC LCO limit. The 300 ppm SR value is sufficiently less negative than the EOC LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met.

SR 3.1.4.~~3~~2 is modified by ~~3~~three Note~~s~~ that include~~s~~ the following requirements:

~~3~~b. If the 300 ppm Surveillance limit is exceeded, it is possible that the EOC limit on MTC could be reached before the planned EOC. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the EOC limit.

~~b~~c. The Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is more positive than the 60 ppm Surveillance limit, the EOC limit will not be exceeded because of the gradual manner in which MTC changes with core burnup.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.

2. FSAR, Chapter [15].

3. WCAP 9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

4. FSAR, Chapter [15].

a. The SR is not required to be performed until 7 effective full power days (EFPDs) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 p.p.m.

New Rev 2

Industry/TSTF Standard Technical Specification Change Traveler

Add an LCO item and SR to Mode 2 Physics Tests Exceptions to verify that Thermal Power <= 5% RTP.

NUREGs Affected: ☐ 1430 ☒ 1431 ☐ 1432 ☐ 1433 ☐ 1434

Description:

Add an LCO requirement and SR to Mode 2 Physics Tests Exceptions 3.1.10 to verify that Thermal Power <= 5% RTP. Deleted references in the Bases to Physics Tests to tests performed in Mode 1 as this Tech Spec only applies to tests performed in Mode 2. Deleted the reference to Mode 2 in the Applicability.

Justification:

This LCO requirements and SR were added to verify that Thermal Power is within the defined power level for Mode 2 during performance of Physics Tests, since there is an action that addresses Thermal Power not within limit and no corresponding LCO or surveillance.
The Bases references to Physics Tests performed in Mode 1 were unnecessary as this specification refers only to tests performed in Mode 2.
The explicit reference to Mode 2 in the Applicability is unnecessary as the LCO limits the use of the Test Exception to power levels less than 5% (the upper limit of Mode 2).

Affected Technical Specifications

3.1.10 Bases	Physics Test Exceptions - Mode 2
LCO 3.1.10	Physics Test Exceptions - Mode 2
LCO 3.1.10 Bases	Physics Test Exceptions - Mode 2
Appl. 3.1.10	Physics Test Exceptions - Mode 2
Appl. 3.1.10 Bases	Physics Test Exceptions - Mode 2
SR 3.1.10.3	Physics Test Exceptions - Mode 2 Change Description: Renumber to 3.1.10.4
SR 3.1.10.3	Physics Test Exceptions - Mode 2 Change Description: Inserted
SR 3.1.10.3 Bases	Physics Test Exceptions - Mode 2 Change Description: Renumber to 3.1.10.4
SR 3.1.10.3 Bases	Physics Test Exceptions - Mode 2 Change Description: Inserted

WOG Review Information

WOG-4.6

Originating Plant: Date Provided to OG: 11-Mar-95 Needed By:

Owners Group History:

WOG-04, C.6

Owners Group Resolution: Approved Date: 11-Aug-95

TSTF Review Information

TSTF Received Date: 05-Sep-95 Date Distributed to OGs for Review: 05-Sep-95

OG Review Completed: ☒ BWOG ☒ WOG ☒ CEOG ☒ BWROG

TSTF History:

TSTF Resolution: Approved Date: 05-Sep-95 TSTF- 14

NRC Review Information

NRC Received Date: 03-Oct-95 NRC Reviewer: R. Tjader Reviewer Phone #:

Reviewer Comments:

10/4/95 - R. Tjader approved change, pkg to TSB mgmt.
11/17/95 - C. Grimes approved change.

Final Resolution: Approved Date: 27-Nov-95

Revision History

Revision 1 Revision Date: 08-Jan-96 Proposed by: TSTF

Revision Description:
Remarked the pages to use TSTF number instead of OG number.
The Tech Spec markup contains other changes not discussed in the Discussion or Justification. The TSTF package was WOG-4, C.6 only, but changes WOG-4, C.1 and C.4 were included in the TSTF package. These were removed.

Resolution: Date:

Revision 2 Revision Date: 15-Jan-96 Proposed by: TSTF

Revision Description:
Added a LCO requirement in addition to the surveillance.

Resolution: Date:

Incorporation Into the NUREGs

File to BBS/LAN Date:

File to TSTF Date:

File Rev Incoporated:

File Rev Incorporated Date

3.1 REACTIVITY CONTROL SYSTEMS

3.1.10 PHYSICS TESTS Exceptions—MODE 2

LCO 3.1.10 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.4, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.5, "Rod Group Alignment Limits";
LCO 3.1.6, "Shutdown Bank Insertion Limits";
LCO 3.1.7, "Control Bank Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended, provided:

a. RCS lowest loop average temperature is ≥ [531]°F; ~~and~~

b. SDM is ≥ [1.6]% $\Delta k/k$; and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY: ~~MODE 2~~ during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. SDM not within limit.	A.1	Initiate boration to restore SDM to within limit.	15 minutes
	AND		
	A.2	Suspend PHYSICS TESTS exceptions.	1 hour
B. THERMAL POWER not within limit.	B.1	Open reactor trip breakers.	Immediately

(continued)

TSTF-14

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.10.1	Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per [SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1].	Within 12 hours prior to initiation of PHYSICS TESTS
SR 3.1.10.2	Verify the RCS lowest loop average temperature is ≥ [531]°F.	30 minutes
SR 3.1.10.~~3~~ 4	Verify SDM is ≥ 1.6% Δk/k.	24 hours
SR 3.1.10.3	Verify THERMAL POWER is ≤ 5% RTP.	30 minutes

BASES

BACKGROUND (continued)

all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

a. Critical Boron Concentration—Control Rods Withdrawn;

b. Critical Boron Concentration—Control Rods Inserted;

c. Control Rod Worth;

d. Isothermal Temperature Coefficient (ITC); and

~~e. Neutron Flux Symmetry.~~

~~The first four tests are performed in MODE 2, and the last test can be performed in either MODE 1 or 2.~~ These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

a. The Critical Boron Concentration—Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical (k_{eff} = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.

b. The Critical Boron Concentration—Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% $\Delta k/k$ when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron

(continued)

BASES

BACKGROUND (continued)

performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

e. The Flux Symmetry Test measures the degree of azimuthal symmetry of the neutron flux at as low a power level as practical, depending on the test method employed. This test can be performed at HZP (Control Rod Worth Symmetry Method) or at ≤ 30% RTP (Flux Distribution Method). The Control Rod Worth Symmetry Method inserts a control bank, which can then be withdrawn to compensate for the insertion of a single control rod from a symmetric set. The symmetric rods of each set are then tested to evaluate the symmetry of the control rod worth and neutron flux (power distribution). A reactivity computer is used to measure the control rod worths. Performance of this test could violate LCO 3.1.5, LCO 3.1.6, or LCO 3.1.7. The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% RTP.

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational

(continued)

TSTF-14

BASES

LCO (continued)

limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ [531] °F; ~~and~~

b. SDM is ≥ [1.6]% Δk/k; and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY

This LCO is applicable in ~~MODE 2~~ when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP. Other PHYSICS TESTS are performed in MODE 1 and are addressed in LCO 3.1.9, "PHYSICS TESTS Exceptions—MODE 1."

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

(continued)

TSTF-14

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.10.2 (continued)

performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

INSERT 1

SR 3.1.10.~~3~~4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. Regulatory Guide 1.68, Revision 2, August, 1978.

4. ANSI/ANS-19.6.1-1985, December 13, 1985.

(continued)

TSTF-14

INSERT 1

SR 3.1.~~8~~10.3

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of [30 minutes] during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
CONVERSION PACKAGE SECTION 3.2

ITS CONVERSION PACKAGE

CHAPTER 3.2 - POWER DISTRIBUTION LIMITS

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.2 - POWER DISTRIBUTION LIMITS

PART 1

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

ITS

A1

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows:

- at positions ≥ 200 steps and is > 15 inches out of alignment with its bank position, or

- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

then within two hours, perform the following:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power

3.10.1.6 Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted.

See 3.1.4 3.1.5 3.1.6 3.1.8

L1

3.10.2 Power Distribution Limits

MODE 1

Applicability]

3.10.2.1 At all times except during low power physics tests, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the Basis, must meet the following limits:

[LCO 3.2.1]

M1

limits specified in the COLR.

LA1

$F_Q(Z) \leq (F_Q^{RTP}/P) \times K(Z)$ for $P > 0.5$

$F_Q(Z) < (F_Q^{RTP}/0.5) \times K(Z)$ for $P \leq 0.5$

as approximated by $F_Q^V(Z)$

LA1

$F_{\Delta H} < F_{\Delta H}^{RTP} (1 + PF_{\Delta H} (1-P))$

See 3.2.2

LA1

See 3.2.2

Add RA A.2.2
RA A.2.4
RA B.1

M2

ITS

A1

where P is the fraction of rated power (2300 Mwt) at which the core is operating. $F_Q(Z)$ is the measured $F_Q^N(Z)$ multiplied by the measurement uncertainty factor $F_u^N = 1.05$ and the engineering factor $F_Q^E = 1.03$. $F_{\Delta H}$ is the measured $F_{\Delta H}^N$ multiplied by a 1.04 measurement uncertainty factor. K(Z) is the normalized $F_Q(Z)$ as a function of core height specified in the CORE OPERATING LIMITS REPORT (COLR). F_Q^{RTP} is the F_Q limit at RATED THERMAL POWER (RTP). $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER. $PF_{\Delta H}$ is the Power Factor Multiplier for $F_{\Delta H}^{RTP}$. F_Q^{RTP}, $F_{\Delta H}^{RTP}$, and $PF_{\Delta H}$ are specified in the COLR.

LA1 | See 3.2.2 | LA1 | See 3.2.2 | LA1 | A2

3.10.2.1.1 [SR 3.2.1.1]

Following ~~initial loading~~ refueling and prior to exceeding 75% RTP, or ~~upon~~ achieving equilibrium conditions after exceeding by 10% or more of rated power, the power $F_Q(Z)$ was last determined, and at least once per ~~effective full power month~~ 31 EFPDs power distribution maps using the ~~movable detector system shall be made to confirm that the hot channel factor~~ $F_Q(Z)$ limits of Specification 3.10.2.1 are satisfied and to establish the target axial flux difference as a function of power level (called the target flux difference).* within 12 hours of

M3 | A3

~~If either measured hot channel factor exceeds the specified limit, the reactor power shall be reduced so as not to exceed a fraction equal to the ratio of the $F_Q(Z)$ or $F_{\Delta H}$ limit to the measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio.~~

See 3.2.3 | A4

[RA A.2.3]

~~If subsequent incore mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met,~~ the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced. 72

See 3.2.2 | L2 | M28

During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

See 3.2.3

ITS

Specification 3.2.1

A1

3.10.2.2 ~~$F_Q(Z)$ shall be determined to be within the limit given in 3.10.2.1 by satisfying the following relationship for the middle axial 80% of the core at the time of the target flux determination:~~

A5

~~$F_Q(Z) \leq (F_Q^{RTP}/P) \times [K(Z)/V(Z)]$ for $P > 0.5$~~

LA1

~~$F_Q(Z) \leq (F_Q^{RTP}/0.5) \times [K(Z)/V(Z)]$ for $P \leq 0.5$~~

~~where $V(Z)$ is specified in the COLR.~~

ITS

Reduce AFD target band limits to restore $F_Q^V(Z)$ to within limit — M4 — A1

3.10.2.2.1 If the relationship specified in 3.10.2.2 cannot be satisfied, one of the following actions shall be taken:

[RA A.1] a) ~~Place the core in an equilibrium condition where the limit in 3.10.2.2 is satisfied and re-establish the target axial flux difference~~ — M4

[RA A.2.1] b) Reduce the reactor power by ~~the maximum percent calculated with the following expression for the middle axial 80% of the core~~: — A6

≥1% RTP for each 1% $F_Q^V(Z)$ exceeds limit — A7

within 30 minutes — M5

$$\left[\left[\text{max. over Z of } \frac{F_Q(Z) \times V(Z)}{(F_Q^{RTP}/P) \times K(Z)}\right] - 1\right] \times 100\%$$

— A6

c) ~~Comply with the requirements of Specification 3.10.2.2.2.~~ — M6

3.10.2.2.2 ~~The Allowable Power Level above which initiation of the Axial Power Distribution Monitoring System (APDMS) is required is given by the relation:~~

$$APL = \text{minimum over Z of } \frac{F_Q^{RTP} \times K(Z)}{F_Q(Z) \times V(Z)} \times 100\%$$

where $F_Q(Z)$ is the measured $F_Q^N(Z)$, multiplied by the engineering factor $F_Q^E = 1.03$ and the measurement uncertainty factor $F_u^N = 1.05$ at the time of target flux determination from a power distribution map using the movable incore detectors. The V(Z) axial variation function and K(Z) functions are specified in the COLR. — See 3.2.3

The above limit is not applicable in the following core plane regions.

1) Lower core region 0% to 10% inclusive.
2) Upper core region 90% to 100% inclusive.

Specification 3.2.1

ITS

A1

At power levels in excess of APL of rated power, the APDMS will be employed to monitor $F_Q(Z)$. The limiting value is expressed as:

$$\left[F_j(Z)\ S(Z) \right]_{max} \leq \frac{\left[F_Q^{RTP} / (F_u^N \times F_Q^E \times F_Q^*)\right]/P}{\bar{R}_j\ (1 + \sigma_j)}$$

M6

where:

a. P is the fraction of rated power (2300 Mwt) at which the core is operating ($P \leq 1.0$).

b. F_u^N = 1.05 is the measurement uncertainty factor.

c. F_Q^E = 1.03 is the engineering uncertainty factor.

d. F_Q^* = 1.02 is the instrument uncertainty factor.

e. $\bar{R}_j$ for thimble j, is determined from the core power maps and is by definition:

$$\bar{R}_j = \frac{1}{6} \sum_{i=1}^{6} \frac{F_{Qi}}{\left[F(Z)_{ij}\ S(Z)\right]_{max}}$$

i) F_{Qi} is the value obtained from a full core map including S(Z), but without the uncertainty factors F_u^N and F_Q^E.

ii) $F(Z)_{ij}\ S(Z)$ is the measured value without inclusion of the instrument uncertainty factor F_Q^*.

f. σ_j is the standard deviation associated with the determination of $\bar{R}_j$.

g. S(Z) is the inverse of the K(Z) function specified in the COLR.

This limit is not applicable during physics tests and excore detector calibrations.

Amendment No. ~~87~~, ~~115~~, ~~119~~, 141

Specification 3.2-1

ITS

A1

30, 60, 120, 180, 240, 360, and 480 minutes following accumulated control rod motion in any one direction of five steps or more, exclusive of control rod movement within 15 steps from the top of the core. From the traverses, determination of F(Z)S(Z) shall be made and shown to result in a value less than the limiting value specified in 3.10.2. If the APDMS is out of service, reactor operation above APL of rated power can be continued for fourteen equivalent full power days provided that traverses are taken manually at equivalent frequencies, and a log of accumulated rod motion and time of manual traverses is kept.

4.11.3 The following criteria will be used for selecting the channels for measuring F(Z)S(Z):

a. The channel is not acceptable if it contains a control rod allowed by the insertion limits at power levels requiring APDMS.

b. For the latest full core power map, i, channels, j, are acceptable if:

$$\frac{R_{ij} - \bar{R}_j}{\bar{R}_j} \leq 2\sigma_j$$

M6

Basis

The $\bar{R}$ technique provides a means for using many of the monitoring thimbles to determine $F_Q(Z)$ without fully mapping the core. Frequent core maps assure that appropriate values of $\bar{R}$ are being used for each thimble.

Upon return to power following a refueling outage or other situation where establishment of normal APDMS operation is required, power operation above APL of rated power is desirable to establish hot channel factors at full power.

A8

ITS

Specification 3.2.1

4.11 REACTOR CORE

A1

APPLICABILITY

Applies to surveillance of the reactor core.

OBJECTIVE

To ensure the integrity of the fuel cladding.

SPECIFICATION

4.11.1 APDMS OPERATION

4.11.1.1 Prior to establishing normal operation with APDMS, at least six maps will be taken to determine applicable values of $\bar{R}$ and σ for surveillance thimbles.

4.11.1.2 Plant operation up to rated power shall be permitted for the purposes of obtaining the initial maps of Specification 4.11.1.1, provided the APDMS is operational and hot channel factors are shown to be below the limiting values set forth in Specification 3.10.2. Suitably conservative values of $\bar{R}$ and σ shall be derived from maps previously run during the current fuel cycle for use in the APDMS system during this initial period.

4.11.1.3 Subsequent updates of $\bar{R}$ and σ shall employ the last six maps in accordance with Specification 4.11.1.1. M6

4.11.1.4 Each power distribution map will be based on flux traverses obtained from 36 or more of the incore thimbles.

4.11.2 Except during physics tests and excore calibrations, axial surveillance of F(Z)S(Z) shall consist of traverses with the movable incore detectors in appropriate pairs of detector paths, taken every eight hours, or a frequency of approximately 0, 10,

ITS

A1

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows:

- at positions ≥ 200 steps and is > 15 inches out of alignment with its bank position, or

- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

then within two hours, perform the following:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power

See 3.1.4 3.1.5 3.1.6 3.1.8

3.10.1.6 Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted.

3.10.2 Power Distribution Limits MODE 1 L3

[LCO 3.2.2] [Applicability]

3.10.2.1 ~~(At all times except during low power physics tests)~~, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the basis, must meet the ~~following limits:~~ M7

Limits Specified in the COLR LA2

$F_Q(Z) \leq (F_Q^{RTP}/P) \times K(Z)$ for $P > 0.5$

$F_Q(Z) < (F_Q^{RTP}/0.5) \times K(Z)$ for $P \leq 0.5$

See 3.2.1

$\cancel{F_{\Delta H} < F_{\Delta H}^{RTP} (1 + PF_{\Delta H} (1-P))}$ LA2

ITS

where P is the fraction of rated power (2300 Mwt) at which the core is operating. $F_Q(Z)$ is the measured $F_Q^N(Z)$ multiplied by the measurement uncertainty factor $F_u^N = 1.05$ and the engineering factor $F_Q^E = 1.03$. $F_{\Delta H}$ is the measured $F_{\Delta H}^N$ multiplied by a 1.04 measurement uncertainty factor. K(Z) is the normalized $F_Q(Z)$ as a function of core height specified in the CORE OPERATING LIMITS REPORT (COLR). F_Q^{RTP} is the F_Q limit at RATED THERMAL POWER (RTP). $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER. $PF_{\Delta H}$ is the Power Factor Multiplier for $F_{\Delta H}^{RTP}$. F_Q^{RTP}, $F_{\Delta H}^{RTP}$, and $PF_{\Delta H}$ are specified in the COLR.

LA2 A1 See 3.2.1 LA2 See 3.2.1 LA2 See 3.2.1 A2

3.10.2.1.1 Following initial loading, or upon achieving equilibrium conditions after exceeding by 10% or more of rated power, the power $F_Q(Z)$ was last determined, and at least once per effective full power month, power distribution maps using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.1 are satisfied and to establish the target axial flux difference as a function of power level (called the target flux difference).*

[SR 3.2.2.1]

3 EFPDs refueling prior to exceeding 75% RTP M8 $F_{\Delta H}$ L4 A3

[RA A.1.1] [RA A.1.2.1] If ~~either measured hot channel factor~~ $F_{\Delta H}$ exceeds the specified limit, the reactor power shall be reduced so as not to exceed a fraction equal to the ratio of the $F_Q(Z)$ or $F_{\Delta H}$ limit to the measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio to ≤ 55% RTP.

See 3.2.3 A3 by 50%

[RA A.1.2.2]

If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

M9 then restore $F_{\Delta H}^N$ to within limits within 4 hours OR M9 See 3.2.1 M10

Add Note to Condition A

	RA	A.2
	RA	A.3
Note to	RA	A.3
	RA	B.1

L5 M9

During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

M11 See 3.2.3

ITS

3.10.2.2.3 With successive measurements indicating the enthalpy rise hot channel factor, $F^N_{\Delta H}$, to be increasing with exposure, the total peaking factor, $F_Q(Z)$, shall be further increased by two percent over that specified in Specifications 3.10.2.2, 3.10.2.2.1, and

[SR 3.2.2.1 NOTE]

$F^V_Q(z)$ A2

ITS

Specification 3.2.2

and reverified to be within limits

M12 A1

[SR 3.2.2.1 NOTE]

3.10.2.2.2 or $F_Q(Z)$ shall be measured and a target axial flux difference re-established at least once every seven (7) effective full power days until two successive measurements indicate enthalpy rise hot channel factor, $F^N_{\Delta H}$, is not increasing.

3.10.2.3 The reference equilibrium-indicated axial flux difference as a function of power level (called the target flux difference) shall be determined in conjunction with the measurement of $F_Q(Z)$ as defined in Specification 3.10.2.1.1.*

3.10.2.4 The indicated axial flux difference shall be considered outside of the limits of Sections 3.10.2.5 through 3.10.2.9 when more than one of the operable excore channels are indicating the axial flux difference to be outside a limit.

3.10.2.5 Except during physics tests, and except as modified by 3.10.2.6 through 3.10.2.9 below, the indicated axial flux difference shall be maintained within the applicable target band about the target flux difference (defines the target band on axial flux difference).

3.10.2.6 At a power level greater than 90 percent of rated power, or 0.9 x APL** (whichever is less), if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band immediately or reactor power shall be reduced to a level no greater than 90 percent of rated power or 0.9 x APL (whichever is less).

3.10.2.7 At a power level between 50 percent and 90 percent of rated power, or 0.9 x APL (whichever is less).

SEE 3.2.3

* During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

** APL is the Allowable Power Level defined in Specification 3.10.2.2.2.

ITS

A1

3.10.2.2.2 or $F_Q(Z)$ shall be measured and a target axial flux difference re-established at least once every seven (7) effective full power days until two successive measurements indicate enthalpy rise hot channel factor, $F^N_{\Delta H}$, is not increasing. See 3.2.2

M13

3.10.2.3 [SR 3.2.3.3 NOTE 2] The reference equilibrium-indicated axial flux difference as a function of power level (called the target flux difference) shall be determined in conjunction with the measurement of $F_Q(Z)$ as defined in Specification 3.10.2.1.1.

M14

3.10.2.4 [LCO 3.2.3.a NOTE] The indicated axial flux difference shall be considered outside of the limits of Sections 3.10.2.5 through 3.10.2.9 when more than one of the operable excore channels are indicating the axial flux difference to be outside a limit.

M15

3.10.2.5 [LCO 3.2.3.a] [CONDITION C] ~~Except during physics tests~~, and except as modified by 3.10.2.6 through 3.10.2.9 below, the indicated axial flux difference shall be maintained within the applicable target band about the target flux difference (defines the target band on axial flux difference) and within acceptable operation limits.

3.10.2.6 [RA A.1] [RA B.1] ~~At a power level greater than 90 percent of rated power~~ or 0.9 x APL (whichever is less), if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band ~~immediately~~ or reactor power shall be reduced to a level ~~no greater~~ less than 90 percent of rated power or 0.9 x APL (whichever is less). within 15 minutes

M16

A9

3.10.2.7 [LCO 3.2.3.b] [CONDITION C] ~~At a power level between 50 percent and 90 percent of rated power~~ or 0.9 x APL (whichever is less). within an additional 15 minutes

M16

M17

M16

APPLICABILITY: MODE 1 WITH THERMAL POWER >15% RTP

RA D.1
SR 3.2.3.1

M18

and within 31 EFPDs

M13

[SR 3.2.3.3 NOTE 1] During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

A21

~~APL is the Allowable Power Level defined in Specification 3.10.2.2.2.~~

Specification 3.2.3

ITS

A1

M15

[LCO 3.2.3.b]
[CONDITION C]
[RA C.1]

a. The indicated axial flux difference may deviate from its target band for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference does not exceed the limits specified in the COLR. If the cumulative time exceeds one hour, then the reactor power shall be reduced ~~immediately~~ to no greater than 50 percent of rated power and the high neutron flux setpoint reduced to no greater than 55 percent of rated power.

Insert 3.2.3-1

within 30 minutes

A10

b. A power increase to a level greater than 90 percent of rated power or 0.9 x APL (whichever is less) is contingent upon the indicated axial flux difference being within its target band.

L6

3.10.2.8 At a power level ~~no greater~~ less than 50 percent of rated power

A11

[LCO 3.2.3c]

a. The indicated axial flux difference may deviate from its target band.

[RA C.2]

b. A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24-hour period. One-half of the time the indicated axial flux difference is out of its target band up to 50 percent of rated power is to be counted as contributing to the one-hour cumulative maximum the flux difference may deviate from its target band at a power level less than or equal to 90 percent of rated power or 0.9 x APL (whichever is less).

M19

[LCO 3.2.3.c NOTE]

3.10.2.9 Calibration of excore detectors will be performed under the following conditions:

[Applicability NOTE]

a. at power levels greater than 90 percent of rated power or 0.9 x APL (whichever is less) provided the axial flux difference does not exceed the specified target bands, or

Insert 3.2.3.2

A12

Insert 3.2.3-3

L7

Add Note 1 to LCO 3.2.3.b

A13

Add Note to Condition C

A14

CTS INSERT 3.2.3-1

". . . the indicated axial flux difference deviates outside of acceptable operation limits or . . ."

CTS INSERT 3.2.3-2

----------NOTE----------
Penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

CTS INSERT 3.2.3-3

----------NOTE----------
A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.6, provided AFD is maintained within acceptable operation limits.

ITS

For each OPERABLE excore channel — A1

b. at power levels less than 90 percent of rated power or 0.9 x APL (whichever is less) provided the indicated axial flux difference does not exceed the limits specified in the COLR. — L7

3.10.2.10 Alarms shall normally be used to indicate non-conformance with the flux difference requirement of 3.10.2.6 or the flux difference-time requirement of 3.10.2.7.a. If the alarms are temporarily out of service, the axial flux difference shall be logged and conformance with the limits assessed, every hour for the first 24 hours, and half-hourly thereafter. — A15

[SR 3.2.3.2 Frequency Note] [SR 3.2.3.2]

Insert 3.2.3.4

3.10.2.11 The axial flux difference target band about the target axial flux difference shall be determined in conjunction with the measurement of $F_Q(Z)$ as specified in 3.10.2.1.1. The allowable values of the target band are specified in the COLR. Redefinition of the target band from more restrictive to less restrictive ranges between determinations of the target axial flux difference is allowed when appropriate redefinitions of APL are made. Redefinition of the target band from less restrictive to more restrictive ranges is allowed only in conjunction with the determination of a new target axial flux difference. — M20, L8, A16

[LCO 3.2.3.a]

3.10.3 Quadrant Power Tilt Limits

3.10.3.1 Except for physics tests and during power increases below 50 percent of rated power, whenever the indicated quadrant power tilt ratio exceeds 1.02, the tilt condition shall be eliminated within two hours or the following actions shall be taken:

a. Restrict core power level and reset the power range high flux setpoint to be less than two percent of rated values for every percent of indicated power tilt ratio exceeding 1.0, and — LA3

See 3.2.4

Add Note to SR 3.2.3.2 — A15

CTS INSERT 3.2.3-4

"Once within 15 minutes and every 15 minutes thereafter when THERMAL POWER ≥ 90% RTP or 0.9 APL, whichever is less

AND

Once within 1 hour and every 1 hour thereafter when THERMAL POWER < 90% RTP or 0.9 APL, whichever is less"

ITS

A1

3.10.2.2.1 If the relationship specified in 3.10.2.2 cannot be satisfied, one of the following actions shall be taken:

a) Place the core in an equilibrium condition where the limit in 3.10.2.2 is satisfied and re-establish the target axial flux difference

b) Reduce the reactor power by the maximum percent calculated with the following expression for the middle axial 80% of the core:

$$\left[\left[\text{max. over Z of } \frac{F_Q(Z) \times V(Z)}{(F_Q^{RTP}/P) \times K(Z)}\right] - 1\right] \times 100\%$$

See 3.2.1

c) Comply with the requirements of Specification 3.10.2.2.2.

3.10.2.2.2 The Allowable Power Level above which initiation of the Axial Power Distribution Monitoring System (APDMS) is required is given by the relation:

[LCO 3.2.3b] Note 2

$$APL = \text{minimum over Z of } \frac{F_Q^{RTP} \times K(Z)}{\cancel{F_Q(Z) \times V(Z)}} \times 100\%$$

$F_Q^V(Z)$

A17

where $F_Q(Z)$ is the measured $F_Q^N(Z)$, multiplied by the engineering factor F_Q^E = 1.03 and the measurement uncertainty factor F_u^N = 1.05 at the time of target flux determination from a power distribution map using the movable incore detectors. The V(Z) axial variation function and K(Z) functions are specified in the COLR.

LA4

The above limit is not applicable in the following core plane regions.

See 3.2.1

1) Lower core region 0% to 10% inclusive.
2) Upper core region 90% to 100% inclusive.

A22

(APL) is the limitation placed on THERMAL POWER for the purposes of applying the AFD target flux and operational limit curves. The APL is

ITS

A1

where P is the fraction of rated power (2300 Mwt) at which the core is operating. $F_Q(Z)$ is the measured $F_Q^N(Z)$ multiplied by the measurement uncertainty factor $F_Q^N = 1.05$ and the engineering factor $F_Q^E = 1.03$. $F_{\Delta H}$ is the measured $F_{\Delta H}^N$ multiplied by a 1.04 measurement uncertainty factor. K(Z) is the normalized $F_Q(Z)$ as a function of core height specified in the CORE OPERATING LIMITS REPORT (COLR). F_Q^{RTP} is the F_Q limit at RATED THERMAL POWER (RTP). $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER. $PF_{\Delta H}$ is the Power Factor Multiplier for $F_{\Delta H}^{RTP}$. F_Q^{RTP}, $F_{\Delta H}^{RTP}$, and $PF_{\Delta H}$ are specified in the COLR.

See 3.2.1 See 3.2.2 See 3.2.1 See 3.2.2 See 3.2.1

A2

3.10.2.1.1 [SR 3.2.3.3]

31 EFPDs

Within 31 EFPDs following refueling

M21

~~Following initial loading, or upon achieving equilibrium conditions after exceeding by 10% or more of rated power, the power $F_Q(Z)$ was last determined, and at least once per effective full power month, power distribution maps using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.1 are satisfied and to establish the target axial flux difference as a function of power level (called the target flux difference).*~~

L9

A18

If either measured hot channel factor exceeds the specified limit, the reactor power shall be reduced so as not to exceed a fraction equal to the ratio of the $F_Q(Z)$ or $F_{\Delta H}$ limit to the measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio.

See 3.2.1 3.2.2

If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

Insert 3.2.3.5

A18

See 3.2.1 3.2.2

and within 31 EFPDs

M13

[SR 3.2.3.1 NOTE 1] During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

CTS INSERT 3.2.3-5

Determine, by measurement, the target flux difference of each OPERABLE excore channel.

ITS

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b. at power levels less than 90 percent of rated power or 0.9 x APL (whichever is less) provided the indicated axial flux difference does not exceed the limits specified in the COLR.

Sec 3.2.3

3.10.2.10 Alarms shall normally be used to indicate non-conformance with the flux difference requirement of 3.10.2.6 or the flux difference-time requirement of 3.10.2.7.a. If the alarms are temporarily out of service, the axial flux difference shall be logged, and conformance with the limits assessed, every hour for the first 24 hours, and half-hourly thereafter.

3.10.2.11 The axial flux difference target band about the target axial flux difference shall be determined in conjunction with the measurement of $F_Q(Z)$ as specified in 3.10.2.1.1. The allowable values of the target band are specified in the COLR. Redefinition of the target band from more restrictive to less restrictive ranges between determinations of the target axial flux difference is allowed when appropriate redefinitions of APL are made. Redefinition of the target band from less restrictive to more restrictive ranges is allowed only in conjunction with the determination of a new target axial flux difference.

M22

3.10.3 Quadrant Power Tilt Limits Ratio

(MODE 1 with THERMAL POWER > 50% RTP)

[applicability]

3.10.3.1 ~~Except for physics tests and during power increases below 50 percent of rated power~~ whenever the indicated quadrant power tilt ratio exceeds 1.02, ~~the tilt condition shall be eliminated within two hours~~ or the following actions shall be taken:

[RA A.1]

L10

(within two hours)

a. Restrict core power level ~~and reset the power range high flux setpoint~~ to be less than ~~two~~ percent of rated values for every percent of indicated power tilt ratio exceeding 1.0, and

M23

three

thermal power

M24

L11

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Add	LCO	3.2.4
	RA	A.2
	RA	A.3
	RA	A.4
	RA	A.5
Note to	RA	A.5
	RA	A.6
Note to	RA	A.6
	SR	3.2.4.1
Note 2 to	SR	3.2.4.1
	SR	3.2.4.2
Note to	SR	3.2.4.2

M26

Specification 3.2.4

ITS

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[RA B.1]

b. If the tilt condition is not eliminated after 24 hours, the power range high flux setpoint shall be reset to 55 percent of rated power. Subsequent reactor operation would be permitted up to 50 percent of rated power for the purpose of measurement and testing to identify the cause of the tilt condition.

3.10.3.2 Except for low power physics tests, if the indicated quadrant tilt exceeds 1.09 and there is simultaneous indication of a misaligned rod:

a. The core power level shall be reduced by 2 percent of rated values for every 1 percent of indicated power tilt exceeding 1.0, and

M25

b. If the tilt condition is not eliminated within two hours, the reactor shall be brought to a hot shutdown condition.

c. After correction of the misaligned rod, reactor operation will be permitted to 50 percent of rated power until the indicated quadrant tilt falls below 1.09.

3.10.3.3 If the indicated quadrant tilt exceeds 1.09 and there is not a simultaneous indication of rod misalignment, except as stated in Specification 3.10.3.2.c, the reactor shall immediately be brought to a hot shutdown condition.

Power shall be reduced to less than 50% within the following 4 hours

L12

L13

ITS

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a. All non-automatic containment isolation valves not required for normal operation are closed and blind flanges are properly installed where required.

b. The equipment door is properly closed and sealed.

c. At least one door in the personnel air lock is properly closed and sealed.

d. All automatic containment isolation trip valves required to be closed during accident conditions are operable or are secured closed except as stated in Specification 3.6.3. Manual valves qualifying as automatic containment isolation valves are secured closed.

e. The uncontrolled containment leakage satisfies Specification 4.4.

See 1.1

1.8 QUADRANT POWER TILT

The quadrant power tilt is defined as the ratio of maximum to average of the upper excore detector currents or the lower excore detector currents, whichever is greater. If one excore is out of service, the three in-service units ~~are~~ can be used in computing the average.

See QPTR Section 1.1

[SR 3.2.4.1] NOTE 1

and reactor power is <75% RTP

M27

1.9 DELETED

1.10 STAGGERED TEST BASIS

A20

A Staggered Test Basis shall consist of:

a. A test schedule for n systems, subsystems, trains or designated components obtained by dividing the specified test interval into n equal subintervals.

See 1.1

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.2 - POWER DISTRIBUTION LIMITS

PART 2

DISCUSSION OF CHANGES (DOCS) FOR CTS MARKUP

ADMINISTRATIVE CHANGES

A1 In the conversion of the H. B. Robinson Steam Electric Plant (HBRSEP), Unit 2 Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)). These changes are administrative, and have no adverse impact on safety.

A2 CTS Specification 3.10.2.1.1 includes the term, "effective full power month," which is changed to 31 Effective Full Power Days (EFPDs) in the ITS to be consistent with NUREG-1431. Both the CTS and ITS terms are equivalent. This change is administrative, and has no adverse impact on safety.

A3 CTS Specification 3.10.2.1.1 applies to both hot channel factors $F_Q(Z)$ and $F_{\Delta H}$. CTS Specification 3.10.2.1.1 is retained in ITS as two Limiting Conditions for Operations (LCOs), which are, ITS Specification 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)" and ITS Specification 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)." As such the term "hot channel factors" and $F_Q(Z)$ in CTS Specification 3.10.2.1.1 is retained as $F^V_Q(Z)$ in ITS Specification 3.2.1 and as $F^N_{\Delta H}$ in ITS Specification 3.2.2. This change is administrative, and has no adverse impact on safety.

A4 CTS Specification 3.10.2.1.1, second paragraph, which contains a required action for the condition where the measured F_Q exceeds the specified limits, is not retained in the ITS. This required action contains a method for reducing power that is less restrictive than CTS Specification 3.10.2.2.1.b, which provides an alternative method that is more conservative than CTS Specification 3.10.2.1.1, second paragraph. CTS Specification 3.10.2.2.1.b requires that the reactor power be reduced by 1% for every 1% that $F_Q(Z)$ exceeds its limits rather than limiting reactor power to the fraction expressed in CTS Specification 3.10.2.1.1 as $F_Q(Z)_{limit}/F^V_Q(Z)_{actual}$. The CTS Specification 3.10.2.2.1.b method of determining the reduced power limitation becomes more conservative than CTS Specification 3.10.2.1.1 as the deviation between the F_Q limits and the measured F_Q increases. CTS Specification 3.10.2.2.1.b is also consistent with NUREG-1431, and is adopted in the ITS. Therefore, this change is administrative, and has no adverse impact on safety.

ADMINISTRATIVE CHANGES
(continued)

A5 CTS Specification 3.10.2.2, first sentence, is redundant to, and refers to, CTS Specification 3.10.2.1, and is not retained in ITS. This change is administrative, and has no adverse impact on safety.

A6 CTS Specifications 3.10.2.2.1.b requires that reactor power be reduced by the expression:

[[max. over Z of $(F_Q(Z) \times V(Z)) / ((F_Q^{RTP}(Z)/P) \times K(Z))] -1] \times 100\%$

when $F_Q^V(Z)$ exceeds the limit. In the bases to ITS, the expression $F_Q(Z) \times V(Z)$ is defined as $F_Q^V(Z)$. In the CORE OPERATING LIMITS REPORT (COLR), the limits for $F_Q^V(Z)$ are defined as $(F_Q^{RTP}(Z)/P) \times K(Z)$. The above expression then reduces to a mathematical equivalent to converting the fraction that $F_Q^V(Z)$ exceeds the limit into a percent RATED THERMAL POWER (RTP). This change is administrative, and has no adverse impact on safety.

A7 CTS Specification 3.10.2.2.1.b, which requires that reactor power be reduced when the measured F_Q exceeds the F_Q limits is retained in ITS Specification 3.2.1 as Required Action A.1.2. CTS Specification 3.10.2.2.1.b also states that the action applies to the ". . . middle axial 80% of the core." This requirement is not retained in ITS, because the axial offset methodology only applies to the middle 80% of the core, as described in the Bases to ITS Specification 3.2.1. This change is therefore administrative, and has no adverse impact on safety.

A8 The CTS Bases are not retained in the ITS, but are replaced in their entirety. The ITS includes significantly expanded and improved Bases. The Bases do not define or impose any specific requirements but serve to explain, clarify and document the reasons (i.e., Bases) for the associated Specification. The Bases are not part of the Technical Specifications required by 10 CFR 50.36. This change is administrative, and has no adverse impact on safety.

A9 CTS Specification 3.10.2.6, which contains the Required Action to return the AXIAL FLUX DIFFERENCE (AFD) to the target band immediately if the AFD is outside of the target band, is modified in the ITS 3.2.3 Required Action A.1 to require a Completion Time of 15 minutes to restore AFD to within the target band. ITS Section 1.3, "Completion Times," states that if "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner. The Completion time of 15 minutes for accomplishing ITS 3.2.3 Required Action A.1 is a reasonable interpretation of the CTS Completion Time of "immediately." Therefore, this change to CTS Specification 3.10.2.6 is administrative, and has no adverse impact on safety.

ADMINISTRATIVE CHANGES
(continued)

A10 CTS Specification 3.10.2.7.a, which contains the Required Action to immediately reduce reactor power to < 50% rated power if cumulative time exceeds one (1) hour if the AFD is outside of the target band, is modified in the ITS 3.2.3 Required Action C.1 to require a Completion Time of 30 minutes to reduce THERMAL POWER to < 50% RTP. ITS Section 1.3, "Completion Times," states that if "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner. The ITS Completion time of 30 minutes for accomplishing Required Action C.1 is a reasonable interpretation of the CTS Completion Time of "immediately," when considering the operating experience associated with reduction in THERMAL POWER from 100% RTP to less than 50% RTP. Therefore, this change to CTS Specification 3.10.2.7.a is administrative, and has no adverse impact on safety.

A11 CTS Specifications 3.10.2.7.b contains requirements that restrict an increase in reactor power above rated power levels in which the particular specifications for AFD apply unless the specifications are met. This requirement duplicates that of CTS 3.10.2.5 which is retained as ITS LCO 3.2.3.a and therefore is not retained in ITS. This change is administrative, and has no adverse impact on safety.

A12 CTS Specification 3.10.2.8.b, which requires the accumulation of penalty deviation time for AFD outside of the target band at power levels less than or equal to 50% reactor power, provides that penalty deviation time be accumulated at one half of the rate that penalty deviation time is accumulated when the reactor is greater than 50% rated power. This requirement is retained in ITS Note to LCO 3.2.3.c, but is completely rewritten for clarity and states that penalty deviation time ". . . shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside of the target band." CTS Specification 3.10.2.8.b states that operation above 50% reactor power is allowed when ". . . the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24-hour period. One-half of the time the indicated AFD is out of its target band up to 50 percent of rated power is to be counted as contributing to the one-hour cumulative maximum the flux difference may deviate from its target band at a power level less than or equal to 90 percent of rated power. . ." The CTS statement is identical in meaning to the ITS Note to LCO 3.2.3.c. When applying the Note to LCO 3.2.3.c to the stated CTS requirement, cumulative penalty hours will add up to a total of one (1) hour for each two (2) hours below 50% RTP. Therefore, this change is administrative, and has no adverse impact on safety.

A13 CTS Specification 3.10.2.7.a, which allows the indicated AFD to deviate from its target band for a maximum of one hour (cumulative) in any 24-

ADMINISTRATIVE CHANGES
(continued)

hour period, is modified in the ITS to add a Note to LCO 3.2.3.b that clarifies the requirement for cumulative penalty time. This change clarifies that cumulative penalty time is accumulated in increments of one minute. Since this change provides clarification only and does not add requirements, this change is administrative, and has no adverse impact on safety.

A14 CTS Specification 3.10.2.7.a, which requires that power be reduced to no greater than 50% rated power, when cumulative penalty time exceeds one hour, is revised in ITS 3.2.3 Required Action C.1 to include a Note to Condition C that Required Action C.1 must be completed whenever Condition C is entered. The CTS does not include ITS LCO 3.0.2, which permits exiting from a Required Action if the LCO is met or no longer applicable prior to the expiration of the specified Completion Time. Hence, the CTS also requires that the required action be completed whenever the Specification requirement is entered. Since the addition of this note only provides clarification with regard to ITS LCO 3.0.2, this change adds no requirements, is administrative, and has no adverse impact on safety.

A15 CTS Specification 3.10.2.10, which requires that alarms shall be normally used to indicate non-conformance with AFD requirements, and if the AFD monitor alarms are out of service, the AFD be logged. This surveillance is retained in ITS and is modified by a Note in ITS Surveillance Requirement (SR) 3.2.3.2 to clarify that logged values should be assumed to exist during the preceding time interval, and by a Note to the SR 3.2.3.2 Frequency that the SR is only required to be performed if the AFD monitor alarm is inoperable. The Note to SR 3.2.3.2 clarifies that the LCO is satisfied for time periods that AFD alarms are operable and not in alarm. This Note is equivalent in meaning to the CTS requirement that alarms ". . . shall normally be used. . ." The Note to SR 3.2.3.2 Frequency clarifies that the LCO is satisfied for the same time periods that AFD alarms are operable and not in alarm, without performance of the SR. This Note is equivalent in meaning to the CTS requirement to log the AFD when the alarms are out of service. Because this change adds clarification and does not add or relax requirements, this change is administrative, and has no adverse impact on safety.

A16 CTS Specification 3.10.2.11, first sentence, which requires that the AFD be determined in conjunction with the measurement of F_Q, is not separately retained in the ITS. The first sentence duplicates the requirements of CTS Specification 3.10.2.3 and CTS Specification 3.10.2.3 is retained in the ITS as SR 3.2.3.3. Therefore, the deletion

ADMINISTRATIVE CHANGES
(continued)

of this duplicate requirement is administrative, and has no adverse impact on safety.

A17 CTS Specification 3.10.2.2.2, which defines the Allowable Power Level (APL) as a function of $F_q(Z)$ and $F_q^V(Z)$, is retained in the ITS as a note to LCO 3.2.3. APL reduces the allowable AFD target as a function of RTP, and therefore is required to ensure that the deviation from target flux difference is within the acceptable target band. The expression $[F_q(Z) \times V(Z)]$ is simplified to the equivalent variable expression $F_q^V(Z)$, which is also defined in the ITS bases. Because this change does not add or reduce requirements, this change is administrative, and has no adverse impact on safety.

A18 CTS Specification 3.10.2.1.1, which requires that power distribution maps using the moveable detector be made to confirm the target AFD is retained in the ITS and restated to "Determine by measurement the target flux difference of each OPERABLE excore channel." The ITS requirement is identical in meaning to the CTS Specification, with the exception that the CTS is silent with respect to whether the AFD is required or not for an inoperable excore channel. Since the target flux difference cannot be determined for inoperable excore channels, this change is administrative, and has no adverse impact on safety.

A19 CTS Specification 3.10.3.1.a, which requires that core power and power range high flux setpoint be reduced when the QUADRANT POWER TILT RATIO (QPTR) is in excess of the limit, is retained in the ITS with the term "rated values" clarified to be "rated thermal power values" to clarify that it is a reduction in rated thermal power that is required when the QPTR limit is exceeded. This is an administrative change, and has no adverse impact on safety.

A20 CTS Specification 1.8, which states that three inservice excore detectors "are" used to determine quadrant power tilt when one is out of service, is revised in ITS SR 3.2.4.1, Note 1, to state that the three remaining power range channels "can be" used for calculating the QPTR. The CTS contains no specific SR Applicability section, and consequently, CTS requirements for surveillance of QPTR when one excore detector is inoperable uses the descriptive verb "are" when describing the surveillance requirement. The ITS includes SR 3.0.1 which, in combination with SR 3.2.4.1 and Note 1, and the ITS definition of QPTR, prohibits determination of QPTR utilizing excore detectors unless three or four excore detectors are OPERABLE. Therefore, the change from "are" to "can be" is administrative, and has no adverse impact on safety.

ADMINISTRATIVE CHANGES
(continued)

A21 The footnote to CTS Specifications 3.10.2.3 and 3.10.2.6, providing a reference for Allowable Power Level (APL) is not retained in ITS. ITS LCO 3.2.3.b, Note 2 adequately defines APL for LCO 3.2.3. This is an administrative change, and has no adverse impact on safety.

A22 CTS Specification 3.10.2.2.2 is revised to add descriptive information for Allowable Power Level (APL) and is retained in ITS LCO 3.2.3.b, Note 2. This is an administrative change, and has no adverse impact on safety.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS Specification 3.10.2.1, which excludes applicability for maintaining F_Q within limits during physics testing, is not retained in ITS. ISTS Specification 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)," does not allow a physics test exception to F_Q limits. The F_Q limits are applicable at all times when the reactor is at power. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M2 The CTS is revised to adopt the Required Actions A.1, A.4 and B.1 from ISTS Specification 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)," as ITS 3.2.1 Required Actions A.2.1, A.2.4 and B.1 in the ITS to ensure that appropriate additional actions are taken when ($F_Q(Z)$ is not within the required limits. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M3 CTS Specification 3.10.2.1.1, which requires that power distribution maps using the movable detector system be made to confirm that the F_Q limits are satisfied following initial loading or upon achieving equilibrium conditions after exceeding by 10% or more of RTP, is retained in ITS as a general Note to the Surveillance Requirements, and in ITS SR 3.2.1.1 has the Frequency changed to refueling interval and prior to exceeding 75% rated power. An additional restriction is imposed in the ITS to perform SR 3.2.1.1 within 12 hours of achieving equilibrium conditions after exceeding by 10% or more of rated power. This change imposes more restrictive requirements, and has no adverse impact on safety.

M4 CTS Specification 3.10.2.2.1.a, which requires that the reactor core be placed in an equilibrium condition where the Heat Flux Hot Channel Factor is satisfied and reestablish the target axial flux difference, is retained and restated in ITS 3.2.3 Required Action A.1. The CTS Required Action, as restated in ITS Required Action A.1, allows the option of reducing the target axial flux band to the ±3% band in order to obtain a lower V(Z) penalty. By restricting operation to the ±3% band rather than the ±5% band, V(Z) is reduced by approximately 3% resulting in a lower $F_Q^V(Z)$. With a lower V(Z) penalty, $F_Q^V(Z)$ may return to within limits without a power reduction. The option provided by Required Action A.1 is consistent with the PDC-3 axial offset control methodology used by Siemens Power Corporation for calculating cycle specific hot channel faction limits (Ref. 1). Because Required Action A.1 imposes the same requirement as CTS 3.10.2.2.1.a, this aspect of the change is administrative.

Also, ITS 3.2.1 Required Action A.1 is revised to include a completion time of 15 minutes for achieving the more restrictive target flux band,

TECHNICAL CHANGES - MORE RESTRICTIVE
(continued)

rather than reestablish the existing target flux band without a required completion time as allowed in the CTS. Since this change imposes the new requirement of a completion time to achieve a more restrictive target band, this change is more restrictive and has no adverse impact on safety.

M5 CTS Specification 3.10.2.2.1.b, which requires that reactor power be reduced if F_Q is not within limits, is retained in ITS Specification 3.2.1 Required Action A.2.1, with the additional requirement of a Completion Time of 30 minutes. The CTS does not impose a required Completion Time. Since this change imposes the new requirement of a completion time to reduce THERMAL POWER, it is more restrictive and has no adverse impact on safety.

M6 CTS Specifications 3.10.2.2.2 defines an APL that permits operation slightly above reduced power levels that are required when hot channel factors, AFD, and QPTR are outside the required limits. Increasing power to the APL requires that the Axial Power Distribution Monitoring System (APDMS) be initiated. This provision in CTS is not retained in ITS. As a result, the ITS Required Actions for reducing power in response to exceeding power distribution limits will be followed without any provision for increasing power to above the ITS Required Action THERMAL POWER levels. Consequently, this change is more restrictive, and has no adverse impact on safety.

In conjunction with this more restrictive change, CTS Specification 4.11, which contains the surveillance requirements for the APDMS, is not retained in ITS. The APDMS is only required to be initiated to support THERMAL POWER levels above those contained in the CTS required actions.

Therefore, these changes are more restrictive, and have no adverse impact on safety.

M7 CTS Specification 3.10.2.1, which excludes applicability for maintaining $F_{\Delta H}$ within limits during physics testing, is not retained in ITS. ISTS Specification 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," does not allow a physics test exception to $F_{\Delta H}$ limits. The $F_{\Delta H}$ limits are applicable at all times when the reactor is at power. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M8 CTS Specification 3.10.2.1.1, which requires that the enthalpy rise hot channel factor, $F_{\Delta H}$, be determined following initial core loading, has the Frequency changed in ITS SR 3.2.2.1 to refueling interval and prior to exceeding 75% RTP. As a result, a limit on THERMAL POWER is imposed

TECHNICAL CHANGES - MORE RESTRICTIVE
(continued)

for the initial performance of SR 3.2.2.1 following a refueling outage. This change imposes more restrictive requirements, and has no adverse impact on safety.

M9 CTS Specification 3.10.2.1.1, second paragraph, requires that the reactor power be reduced in the event that $F_{\Delta H}$ is not within limits. CTS Specification 3.10.2.1.1, second paragraph, is retained in ITS 3.2.2 as Required Actions A.1.1 and A.1.2. No completion time is required in the CTS for the required action. A Completion Time of 4 hours is imposed in the ITS for Required Action A.1.1 and A.1.2. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M10 CTS Specification 3.10.2.1.1, second paragraph, includes the requirement that reactor power be limited to the fraction of RTP equal to $F_{\Delta H}^{limit}/F_{\Delta H}^{actual}$, and that the high neutron flux setpoint be reduced by the same ratio. ITS 3.2.2 Required Action A.1.2 requires that THERMAL POWER be reduced to less than 50% RTP, and that the Power Range Neutron Flux high setpoint be reduced to ≤ 55% RTP. The ITS requirement to reduce to below 50% RTP is more restrictive for values of $F_{\Delta H}$ in excess of the limits up to twice the required limits. Since the Surveillance Frequency is sufficiently short that any $F_{\Delta H}$ measurement in excess of limits is reasonably assured to be less than twice the required limits, this change is considered more restrictive, and has no adverse impact on safety.

M11 The CTS is revised in the ITS to adopt a Note to Condition A, Required Actions A.1.2.1, A.2, A.3 and Note, and B.1, from ISTS 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^{N}$)," when $F_{\Delta H}^{N}$ is not within limits. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M12 CTS Specification 3.10.2.2.2, which imposes additional requirements upon $F_Q(Z)$ (i.e., increases the measured $F_Q(Z)$ in the direction of the limit) if the enthalpy rise hot channel factor is increasing, is retained in ITS Surveillance Requirement 3.2.2.1 as a Note, and is further revised to ensure that $F_Q(Z)$ is reverified to be within the required F_Q limits. While the CTS requirement to remain within F_Q limits remains unchanged, the additional requirement to reverify that $F_Q(Z)$ is within the F_Q limits adds new requirements. Therefore, this change is more restrictive, and has no adverse impact on safety.

M13 CTS Specification 3.10.2.1.1, which requires that the target AFD be established following initial loading, includes a footnote that allows the "design target value" to be used during power escalation until

TECHNICAL CHANGES - MORE RESTRICTIVE
(continued)

extended operation is achieved and the target values can be determined from actual core parameters. The footnote to CTS Specification 3.10.2.1.1 does not include a specific Completion Time upon which the target flux difference must be established based on actual core parameters. The ITS requires that the target flux difference be initially determined within 31 EFPDs of refueling. This change imposes more restrictive requirements, and has no adverse impact on safety.

M14 CTS Specification 3.10.2.5, which excludes applicability for maintaining AFD within the target band during physics testing, is not retained in ITS. ISTS Specification 3.2.3, "Axial Flux Difference (AFD) (PDC-3 Axial Offset Control Methodology)," does not allow a physics test exception for AFD Applicability. The AFD must be maintained as specified by LCO 3.2.3 at all times when the reactor is at power. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M15 CTS Specification 3.10.2.5, which requires that the AFD be maintained within its target band, is revised in ITS LCO 3.2.3.b to also require that the AFD be within the acceptable operation limits. Similarly, CTS Specification 3.10.2.7.a, which requires actions to be taken if AFD is outside of its target band, is revised in ITS 3.2.3 Condition C.1 to apply Required Action C.1 when the AFD is outside of acceptable operation limits. Since this change adds requirements, this change is more restrictive, and has no adverse impact on safety.

M16 CTS Specification 3.10.2.6, which requires actions to be taken to maintain the AFD within the target band for rated power greater than 90% of rated power or 0.9 APL (whichever is less), and CTS Specification 3.10.2.7, which defines actions that result in accumulation of penalty deviation time when the reactor power is ≥ 50% rated power, and less than 90% rated power or 0.9 APL (whichever is less), are revised in ITS 3.2.3 Applicability to MODE 1 with THERMAL POWER > 15% RTP. Since this change imposes applicability for THERMAL POWER < 50%, this change is more restrictive, and has no adverse impact on safety.

The CTS requirement to maintain AFD to within the target band at a THERMAL POWER > 90% RTP or 0.9 APL (whichever is less), is revised in Required Action A.1 for THERMAL POWER ≥ 90% RTP or 0.9 APL (whichever is less). Because this change could potentially result in remaining outside the target band and accumulation of penalty deviation time at exactly 90% RTP or 0.9 APL (whichever is less) or could result in the reduction of THERMAL POWER to below 90% RTP or 0.9 APL (whichever is less) rather than ≤ 90% RTP or 0.9 APL (whichever is less), this change is more restrictive, and has no adverse impact on safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
(continued)

Similarly, the CTS requirement to accumulate penalty deviation time at THERMAL POWER > 50% RTP and < 90% RTP or 0.9 APL (whichever is less), is revised in ITS to define the allowable range for cumulative penalty time to be THERMAL POWER ≥ 50% RTP and < 90% RTP or 0.9 APL (whichever is less). Because this change could result in the accumulation of penalty deviation time at exactly 50% RTP at the rate defined in LCO 3.2.3.b, this change is more restrictive, and has no adverse impact on safety.

M17 CTS Specification 3.10.2.6, which contains the Required Action to return the AFD to the target band or reduce reactor power to less than 90% rated power, is modified in the ITS to require a Completion Time of 15 minutes to reduce THERMAL POWER to < 90% RTP, if the Required Action and associated Completion Time for restoration of AFD to within its target band is not met. No Completion Time for reduction of THERMAL POWER is required in the CTS. The addition of a Completion Time of 15 minutes to reduce power in ITS Required Action B.1 imposes new requirements; therefore, this change is more restrictive and has no adverse impact on safety.

M18 The CTS is revised in the ITS to add a Note to Condition D, add Required Action D.1, and add SR 3.2.3.1, from ISTS 3.2.3, "Axial Flux Difference (AFD) (PDC-3 Axial Offset Control Methodology)." CTS Specification 3.10.2.1.1 does not specifically apply when reactor power is less than 50% rated power except for the purposes of accumulating penalty hours. Required Action D.1 requires that THERMAL POWER be reduced to < 15% RTP if the Required Actions and Completion Times of Condition C are not met. The CTS has no explicit required action if the Required Actions equivalent to Condition C are not met. ITS SR 3.2.3.1 requires verification that AFD is within limits for each OPERABLE excore channel every 7 days. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M19 CTS Specification 3.10.2.8.a, which allows the indicated AFD to deviate from its target band at reactor power ≤ 50% rated power, is revised in the ITS to allow AFD to deviate outside the target band with THERMAL POWER < 50% RTP. Since this change does not allow operation with AFD outside of the target band at exactly 50% RTP, this change imposes more restrictive requirements. and has no adverse impact on safety.

M20 CTS Specification 3.10.2.10, which requires that the AFD be logged every hour for the first 24 hours, and half-hourly thereafter, when the AFD alarm is out of service, is revised in ITS SR 3.2.3.2 to have a Frequency of once within 15 minutes and every 15 minutes thereafter when THERMAL POWER is ≥ 90% RTP, and once within 1 hour and every 1 hour thereafter when THERMAL Power is < 90 % RTP. This change is more restrictive in the case where THERMAL POWER ≥ 90% RTP, without regard to

TECHNICAL CHANGES - MORE RESTRICTIVE
(continued)

how long the AFD monitor has been out of service. Therefore, this change has no adverse impact on safety.

M21 CTS Specification 3.10.2.1.1, which requires that the target AFD as a function of power level be established following initial core loading, has the Frequency changed in SR 3.2.3.3 of the ITS to within 31 EFPDs following each refueling. This change imposes a time limit in the Frequency for the initial performance of SR 3.2.3.3 after refueling. This change imposes more restrictive requirements, and has no adverse impact on safety.

M22 CTS Specification 3.10.3.1, which excludes applicability for required actions when QPTR exceeds 1.02 during physics testing, is not retained in ITS. ISTS Specification 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," does not allow a physics test exception for QPTR applicability. The QPTR must be maintained as specified by LCO 3.2.4 in MODE 1 with THERMAL POWER ≥ 50% RTP. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M23 CTS Specification 3.10.3.1, which requires that actions be taken when QPTR exceeds the limit of 1.02, includes the required action that ". . . the tilt condition shall be eliminated within two hours. . ." This required action, which can be taken in lieu of other required actions that result in a reduction in THERMAL POWER, is not retained in ITS. Because the CTS required action allows two hours to lapse prior to applying a required action to reduce power, this change is more restrictive and has no adverse impact on safety.

M24 CTS Specification 3.10.3.1.a, which requires that power level be reduced in response to QPTR in excess of limit, is revised in the ITS to require a Completion Time of 2 hours to achieve the reduction in THERMAL POWER. Since this change adds a Completion Time requirement, this change is more restrictive, and has no impact on safety.

M25 CTS Specification 3.10.3.1.a, which requires reactor power to be reduced by more than two (2) percent of rated reactor power for every one (1) percent that the QPTR is in excess of the limit, is retained in ITS 3.2.4 Required Action A.1 and is revised to specify a Completion Time of 2 hours and to reduce power by three (3) percent for every percent of QPTR in excess of the limit. Since this change adds a Completion Time for the Required Action which did not exist previously, and restricts THERMAL POWER further as revised in the Required Action, this change is more restrictive, and has no adverse impact on safety.

Similarly, CTS Specification 3.10.3.2, which requires reactor power to be reduced by more than two (2) percent of rated thermal power for every

TECHNICAL CHANGES - MORE RESTRICTIVE
(continued)

one (1) percent of indicated power tilt, is revised in ITS 3.2.4 Required Action A.1 to reduce power by three (3) percent for every percent of QPTR in excess of the limit. This change is more restrictive, and has no adverse impact on safety.

M26 The CTS is revised in the ITS to add a specific LCO to maintain QPTR ≤ 1.02, and add Required Actions A.2, A.3, A.4, A.5 and Note, A.6 and Note, SR 3.2.4.1 and Note 2, and SR 3.2.4.2 and Note from ISTS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)." Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M27 CTS Specification 1.8 allows calculation of QPTR with only three (3) operable power range nuclear instruments without restrictions on reactor power. This specification is retained as Note 1 to ITS SR 3.2.4.1 with the additional restriction from ISTS SR 3.2.4.1 that THERMAL POWER must be < 75% RTP prior to determining QPTR with only three operable excore detectors. When THERMAL POWER is ≥ 75% RTP and one power range nuclear instrument is inoperable, QPTR is determined using the Incore Flux Mapping System. Since this change adds a restriction for THERMAL POWER levels when performing a surveillance under certain conditions, this change is more restrictive, and has no inverse impact on safety.

M28 CTS Specification 3.10.2.1.1, which requires that the Overpower Delta-Temperature (OPΔT) and Overtemperature Delta-Temperature (OTΔT) trip setpoints be reduced if ". . .subsequent incore mapping cannot . . . demonstrate that the hot channel factors are met." is retained in ITS 3.2.1 Required Action A.2.3 to reduce the OPΔT and OTΔT setpoints. However, Required Action A.2.3 must be followed in ITS regardless of the means by which the hot channel factors are measured, i.e., "subsequent incore mapping." Because the ITS Required Action is not restricted by the method used for hot channel factor measurement, this change is considered more restrictive, and has no adverse impact on safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The details contained in CTS Specifications 3.10.2.1, and 3.10.2.2 related to the power distribution limits of $F_Q(Z)$, are relocated to the COLR. This detail, which includes the mathematical relationship of the F_Q, i.e., $F_Q(Z)$, to the normalized hot channel factor, i.e., K(Z), as a function of power, and the associated engineering uncertainty factors, is not required to be in the ITS to provide adequate protection of the health and safety of the public, since the ITS still retains the requirement to remain within the hot channel factor limits specified in the COLR. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable.

LA2 The details contained in CTS Specifications 3.10.2.1, related to the power distribution limits of $F^N_{\Delta H}$, are relocated to the COLR. This detail, and the associated engineering uncertainty factors, is not required to be in the ITS to provide adequate protection of the health and safety of the public, since the ITS still retains the requirement to remain within the hot channel factor limits specified in the COLR. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable.

LA3 The details contained in CTS Specification 3.10.2.11, third and fourth sentences, related to the redefinition of the target band between the less restrictive and the more restrictive ranges, are relocated to the COLR. This detail, which redefines the target band from the more restrictive to the less restrictive range for AFD, is not required to be in the ITS to provide adequate protection of the health and safety of the public, since the ITS still retains the requirement to remain within the target band specified in the COLR. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)
(continued)

LA4 The details contained in CTS Specification 3.10.2.2.2, that define the variable expression $F_Q(Z)$ as the measured hot channel factor, are relocated to the ITS bases. This detail, is not required to be in the ITS to provide adequate protection of the health and safety of the public, since the ITS still retains the requirement to remain within the limits of $F_Q(Z)$. Changes to the ITS bases are controlled in accordance with the ITS Section 5.5.14, "Technical Specifications (TS) Bases Control Program." This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS Specification 3.10.2.1, requires that the F_Q limits be applicable at all times except during physics testing, is revised in ITS Specification 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)," to require that the F_Q limits be applicable in MODE 1 only, and is less restrictive. This change is acceptable, however, since it is only in MODE 1 that a challenge to the F_Q limits can be made. This change does not reduce any margins to safety and is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L2 CTS Specification 3.10.2.1.1, which requires that if the hot channel factors cannot be returned to within limits within 24 hours then the OPΔT and OTΔT setpoints will be reduced by a fraction equal to $F_Q(Z)_{limit}/F_Q^V(Z)_{actual}$, is revised in ITS Specification 3.2.1, "Power Distribution Limits," Required Action A.2.3, to require that if F_Q cannot be returned to within limits within 72 hours the OPΔT and OTΔT setpoints will be reduced. This is a relaxation of requirements, and is less restrictive. This change is acceptable because appropriate time is needed to change the OTΔT and OPΔT setpoints; the 72 hour time period permits the possibility of restoring hot channel factors within limits and may avoid resetting the OTΔT and OPΔT setpoints twice while in Condition A; and, THERMAL POWER has already been reduced to ensure that the hot channel factors are within limits during the time that the plant remains in Condition A. This change is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L3 CTS Specification 3.10.2.1.1, which requires that the $F_{\Delta H}$ limits be applicable at all times except during physics testing, is revised in ITS Specification 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)," to require that the $F_{\Delta H}$ limits be applicable in MODE 1 only, which is less restrictive with respect to applicability to MODEs other than MODE 1. This change is acceptable, however, since it is only in MODE 1 that sufficient THERMAL POWER occurs that could result in a challenge to the $F_{\Delta H}$ limits. This change does not reduce any margins to safety and is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L4 CTS Specification 3.10.2.1.1 contains a surveillance requirement that $F_{\Delta H}$ be verified after exceeding by 10% the power level at which $F_Q(Z)$ was last determined once equilibrium conditions are established following refueling. This surveillance requirement is retained in ITS as SR 3.2.2.1 with the Frequency requirement that $F_{\Delta H}$ be verified prior to exceeding 75% RTP following refueling, and once per 31 EFPDs thereafter, but without the additional restriction of verifying $F_{\Delta H}$ after exceeding by 10% the power level at which F_Q was last measured. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, since further reconfirmation of $F_{\Delta H}$ in

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)
(continued)

addition to the Frequency stated in ITS SR 3.2.2.1 is unnecessary. The measurement of $F_{\Delta H}$ is a function of fuel burnup and is relatively insensitive to changes in reactor power. This change is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L5 CTS Specification 3.10.2.1.1, second and third paragraph, which requires that the OTΔT and OPΔT setpoints be reduced by the fraction $F_{\Delta Hlimit}/F_{\Delta Hactual}$ if the out of limit condition for $F_{\Delta H}$ is not corrected within 24 hours, is not retained in ITS. This is a relaxation of requirements, and is less restrictive. This change is acceptable, since the Required Action to reduce THERMAL POWER to below 50% will likely result in an enthalpy rise hot channel factor that is well below the limiting value at this power level. Further reduction of the OPΔT and OTΔT setpoints is a small contribution to the safety margin, i.e., a OPΔT or OTΔT trip could potentially occur at the reduced setpoint prior to a high neutron flux trip at 55% RTP in response to a transient. While the earlier OPΔT or OTΔT trip could result in a slight improvement in safety margin, this contribution to the safety margin is expected to be small. This change is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology, and has no adverse impact on safety.

L6 CTS Specification 3.10.2.7.a, which requires that in the event that the cumulative penalty time for AFD outside the target band exceeds one hour, the high neutron flux setpoint be reduced to no greater than 55% of rated power, is not retained in the ITS. This is a relaxation of requirements, and is less restrictive. This change is acceptable because Required Action C.1 assures that the plant remains within analyzed parameters for AFD by reducing power and thereby adding margin for AFD to the analyzed assumptions; Required Action D.1 assures that if Required Action C.1 cannot be met within the Completion Time, reactor power is further reduced to add additional margin for AFD to the analyzed assumptions; lowering the high neutron flux setpoints as an additional action does not add appreciable margin to the AFD assumptions in the accident analyses; and, the lower high flux setpoints are not included in the safety analysis assumptions. This change is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L7 CTS Specification 3.10.2.9, which allows calibration of the excore detectors if the AFD is not outside of the target band for > 90% rated power, and if the AFD does not exceed the limits specified in the COLR for reactor power between 50% and 90% rated power, is revised in the ITS Note to Applicability for LCO 3.2.3 to allow up to 16 hours to be

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)
(continued)

accumulated with AFD outside of the target band without penalty deviation time while the excore detectors are being calibrated. This is a relaxation of requirements, and is less restrictive. This change is acceptable because some deviation from the target band is necessary to perform the calibration, and the axial offsets that are used to calibrate the excore detectors alternate between a plus and minus axial offset, such that the overall effect on axial xenon distribution is small. This change is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L8 CTS Specification 3.10.2.10, which requires that the AFD be logged every hour for the first 24 hours, and half-hourly thereafter, when the AFD alarm is out of service, is revised in ITS SR 3.2.3.2 to have a Frequency of once within 15 minutes and every 15 minutes thereafter when THERMAL POWER is ≥ 90% RTP, and once within 1 hour and every 1 hour thereafter when THERMAL POWER is < 90 % RTP. This change is less restrictive in the case that the AFD monitor alarm remains out of service for greater than 24 hours and THERMAL POWER < 90% RTP. This change is acceptable because the likelihood of AFD being out of the target band decreases as steady state operation continues; and, AFD is also more likely to remain within the target band with THERMAL POWER < 90%. This change is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L9 CTS Specification 3.10.2.1.1 contains a surveillance requirement that the target AFD be established after exceeding by 10% the power level at which F_Q was last determined once equilibrium conditions are established following refueling. This surveillance requirement is retained in ITS as SR 3.2.3.3 with the Frequency requirement that the target AFD be established prior to exceeding 75% RTP following refueling, and once per 31 EFPDs thereafter, but without the additional restriction of establishing the target again after exceeding by 10% the power level at which F_Q was last measured. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, since determination of the target AFD is adequately addressed in the PDC-3 axial offset control methodology and is reflected in the requirements stated in ITS LCO 3.2.3. This change is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L10 CTS Specification 3.10.3.1, which excludes applicability for required actions when QPTR exceeds the limit is retained in ITS as an Applicability of MODE 1 with THERMAL POWER > 50% RTP. Since the restated applicability excludes the CTS required applicability for QPTR of exactly 50% RTP this change is considered less restrictive. This change is acceptable since the likelihood of a quadrant power tilt in

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)
(continued)

excess of the limit at exactly 50% RTP resulting in an unanalyzed condition is very small. This change is consistent with NUREG-1431.

L11 CTS Specification 3.10.3.1.a, which requires that the power range high flux setpoint be reset by two (2) percent for every percent that QPTR exceeds 1.0, is not retained in the ITS. Because this change eliminates a requirement, this change is less restrictive. This change is acceptable because the Required Actions remaining in the ITS result in an appropriate reduction in THERMAL POWER to maintain the required safety margins when QPTR is in excess of the limit. This change is consistent with NUREG-1431.

L12 CTS Specification 3.10.3.1.b, which requires that reactor power be reduced to 50% rated power and the power range high flux setpoint reset to 55%, if QPTR is not eliminated within 24 hours, is revised as Required Action B.1 to ITS LCO 3.2.4, This change is less restrictive for two reasons. First, the addition of Required Actions A.2, A.3, A.4, A.5, and A.6, result in the possibility of continued plant operation above 50% RTP with QPTR in excess of the limit as long as the required power reductions are maintained, the F_Q and $F_{\Delta H}$ limits are maintained, and the QPTR condition remains analyzed for the duration of the cycle. CTS Specification 3.10.3.1.b has no such provisions to allow operation above 50% power if the quadrant power tilt remains for more than 24 hours. This change is acceptable because the Required Actions added to Condition A result in the plant remaining in an analyzed condition when the Required Actions are satisfied. Secondly, the requirement to reset the power range high flux setpoints to 55% power is not retained in Required Action B.1. This change is acceptable because the Required Actions remaining in the ITS result in an appropriate reduction in THERMAL POWER to maintain the required safety margins when QPTR is in excess of the limit. This change is consistent with NUREG-1431.

L13 CTS Specifications 3.10.3.2 and 3.10.3.3, which restrict operation with the QPTR in excess of 1.09, is not retained in the ITS. The required actions could potentially result in transition to MODE 3. These restrictions are not retained in the ITS to be consistent with NUREG-1431 which defines required actions specific to the individual LCOs and does not mix required actions from different LCOs. Additionally, the Required Actions to LCO 3.2.4 can result in operation at reduced THERMAL POWER levels. This change is a relaxation of requirements and is less restrictive because plant operation may continue in MODE 1 with QPTR > 1.09 if Required Actions to ITS LCO 3.2.4 were met. This change is acceptable because operation of the plant in accordance with the Required Actions of ITS LCO 3.2.4 reasonably assures that plant operations are within the bounds of the safety analysis.

REFERENCES

1. ANF-88-054(P), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," Advanced Nuclear Fuels Corporation, Richland, WA 99352 (Submitted to NRC by CP&L letter dated August 24, 1989, Proprietary).

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.2 - POWER DISTRIBUTION LIMITS

PART 3

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC), AND BASIS FOR CATEGORICAL EXCLUSION FROM 10 CFR 51.22

ADMINISTRATIVE CHANGES
("A" Labeled Comments/Discussions)

In the conversion of the HBRSEP Unit No. 2 Technical Specifications to the proposed plant specific Improved Technical Specifications certain wording preferences or conventions are being adopted which do not result in technical changes (either actual or interpretational). Editorial changes, clarification, reformatting, rewording and revised numbering are being adopted to make the improved Technical Specifications consistent with NUREG 1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Administrative" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes consist of editorial changes and clarification, reformatting, rewording and renumbering of the current Technical Specifications. This process does not involve any technical changes to existing requirements. As such, these changes are administrative in nature and do not impact initiators of analyzed events or alter any assumptions relative to mitigation of accident or transient events. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. The proposed changes do not impose or eliminate any requirements. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed changes will not reduce a margin of safety because they do not impact any safety analysis assumptions. These changes are administrative in nature and, as such, do not impact any technical requirements. Therefore, these changes do not involve any reduction in a margin of safety.

MORE RESTRICTIVE CHANGES
("M" Labeled Comments/Discussions)

The HBRSEP Unit No. 2 Technical Specifications are proposed to be modified in some areas to impose more restrictive requirements than currently exist. These more restrictive changes are being imposed to be consistent with NUREG 1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "More Restrictive" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes provide requirements determined to be more restrictive than the current Technical Specifications requirements for operation of the facility. These more restrictive requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. These changes have been confirmed to ensure that no previously evaluated accident has been adversely affected. The more restrictive requirements being proposed enhance assurance that process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis of the unit. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. These changes do impose new or additional requirements which are consistent with assumptions made in the safety analysis and licensing basis. The additional requirements include new Surveillance Requirements, more restrictive Frequencies and Completion Times, new LCOs, more restrictive Required Actions and Applicabilities, and other operational restrictions that enhance safe operation. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact or increases the margin of plant safety. Each of the changes in this category, while providing new or additional requirements designed to

enhance plant safety, is consistent with the safety analyses and licensing basis. Therefore, these changes do not involve a reduction in a margin of safety.

LESS RESTRICTIVE-GENERIC CHANGES
("LA" Labeled Comments/Discussions)

In the conversion of the HBRSEP Unit No. 2 Technical Specifications to the proposed plant specific Improved Technical Specifications, portions of some Specifications which are descriptive in nature regarding equipment, systems, actions, surveillances or programs are proposed to be relocated from the Specifications to the Bases, Updated Final Safety Analysis Report, procedures or other licensee controlled documents. The details associated with the involved specifications are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for compliance with the applicable specifications. Changes to the Bases are controlled in accordance with the proposed Bases Control Program described in Chapter 5 of the Improved Technical Specifications. Changes to the UFSAR and administrative procedures which control revisions to these relocated requirements are controlled in accordance with licensee controlled programs.

This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Less Restrictive-Generic" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes relocate requirements from the Technical Specifications to the Bases, Updated Final Safety Analysis Report, procedures or other licensee controlled documents. The documents containing the relocated requirements are subject to the change control of licensee controlled programs. Since any changes to these documents will be evaluated in accordance with the requirements of licensee controlled programs, no increase in the probability or consequences of an accident previously evaluated will be permitted without further NRC review. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. These changes do not introduce a new mode of plant operation. Since any future changes to these requirements will be evaluated in accordance with licensee controlled programs, the possibility of a new or different kind of accident from any accident previously evaluated will not be permitted without further NRC review. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed changes will not reduce a margin of safety because they do not impact any safety analysis assumptions. The requirements that are transposed from the Technical Specifications to other licensee controlled documents are the same as the existing Technical Specifications. Since any future changes to these requirements will be evaluated in accordance with the requirements of licensee controlled programs, no reduction in any margin of safety will be permitted without further NRC review. Therefore, these changes do not involve any reduction in a margin of safety.

LESS RESTRICTIVE SPECIFIC CHANGES
("Lx" Labeled Comments/Discussions)

L1 Change

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The F_Q limits are not a parameter that results in the initiation of an accident, and the F_Q limits can only be challenged in MODE 1. The probability and consequences of an accident occurring is independent of the status of the F_Q limits in Modes other than MODE 1. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The change will continue to require that the F_Q limits be observed during MODE 1. The proposed change does not introduce a new mode of operation or changes in the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

The elimination of applicability of the F_Q limits from modes other than MODE 1 is acceptable since the F_Q limits can only be challenged during MODE 1 operation. The F_Q limits are a steady state parameter that is assumed to be within the limit for accident analyses, and this assumption includes the analyzed safety margin. As THERMAL POWER is reduced below 100% RATED THERMAL POWER (RTP) the safety margin for F_Q increases well above the analyzed safety margin. Since MODE 1 applicability continues from 100% RTP down to 5% RTP, the safety margin for F_Q with the plant in MODE 2 is significantly larger than that assumed in the safety analyses.

Therefore, the elimination of applicability of the F_Q limits for Modes other than MODE 1 does not decrease the margin of safety.

L2 Change

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change increases the time permitted to reduce the Overpower Delta-Temperature (OPΔT) and Overtemperature Delta-Temperature (OTΔT) setpoints from 24 hours to 72 hours in the event that F_Q cannot be restored to within the limits within 24 hours. The probability of occurrence of an accident is independent of the setpoints for protective functions. The consequences of an event during the additional time included in the proposed change is the same as the consequences of an event occurring during the 24 hours currently permitted. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The possibility of a new or different kind of accident is independent of the time permitted to reduce the OPΔT and OTΔT setpoints. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will delay the reduction of the OPΔT and OTΔT setpoints for an additional 48 hours. The measure of safety provided by the reduction in THERMAL POWER is increased slightly, however, when the OPΔT and OTΔT setpoints are reduced, because the OPΔT and OTΔT setpoints may act sooner to trip the plant in response to an accident. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1 that provides the greatest measure of safety to compensate for this condition. Therefore, this change does not involve a significant reduction in a margin of safety.

L3 Change

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The $F_{\Delta H}$ limits are not a parameter that results in the initiation of an accident, and the $F_{\Delta H}$ limits can only be challenged in MODE 1. The probability and consequences of an accident occurring is independent of the status of the $F_{\Delta H}$ limits in Modes other than MODE 1. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The change will continue to require that the $F_{\Delta H}$ limits be observed during MODE 1. The proposed change does not introduce a new mode of operation or changes in the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

The elimination of applicability of the $F_{\Delta H}$ limits from modes other than MODE 1 is acceptable since it is only in MODE 1 that sufficient THERMAL POWER exists that the $F_{\Delta H}$ limits can be challenged. The $F_{\Delta H}$ limits are a steady state parameter that is assumed to be within the limits for accident analyses, and this assumption includes the analyzed safety margin. As THERMAL POWER is reduced below 100% RTP the safety margin for $F_{\Delta H}$ increases well above the analyzed safety margin. Since MODE 1 applicability continues from 100% RTP down to 5% RTP, the safety margin for $F_{\Delta H}$ with the plant in MODE 2 is significantly larger than that assumed in the safety analyses.

Therefore, the elimination of applicability of the $F_{\Delta H}$ limits for Modes other than MODE 1 does not decrease the margin of safety.

L4 Change

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. This change eliminates a Surveillance Frequency for $F_{\Delta H}$ when THERMAL POWER is changed by 10% from the level at which F_Q was last measured. The probability of accidents previously evaluated are independent of Surveillance Frequency for $F_{\Delta H}$. $F_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. Only bank insertion is related to changes in THERMAL POWER of 10% or more. ITS Limiting Condition for Operations (LCO) 3.1.6, "Control Bank Insertion Limits," protects the fuel in part and ensures that the initial conditions assumed in the accident analyses are valid. Because $F_{\Delta H}$ is a slowly changing

parameter, the ITS Surveillance Frequency of once after each refueling prior to THERMAL POWER exceeding 75% RTP and 31 Effective Full Power Days (EFPDs) thereafter provides reasonable assurance that the $F_{\Delta H}$ will not be exceeded. Therefore, this change does not involve an increase in the consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The change is a reduction in Surveillance Frequency. The frequency of surveillance does not create the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   This change eliminates a Surveillance Frequency for $F_{\Delta H}$ when THERMAL POWER is changed by 10% from the level at which F_Q was last measured. $F_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. Only bank insertion is related to changes in THERMAL POWER of 10% or more. ITS LCO 3.1.6, "Control Bank Insertion Limits," protects the fuel in part and ensures that the initial conditions assumed in the accident analyses are valid. Because $F_{\Delta H}$ is a slowly changing parameter, the ITS Surveillance Frequency of once after each refueling prior to THERMAL POWER exceeding 75% RTP and 31 EFPD thereafter is sufficient to assure that $F_{\Delta H}$ will not exceed its limits for a significant period of time. Therefore, this change does not involve a significant reduction in a margin of safety.

L5 Change

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. This change eliminates a requirement to reduce the OTΔT and OPΔT setpoints by the fraction $F_{\Delta H limit}/F_{\Delta H actual}$ if the out of limit condition for $F_{\Delta H}$ is not corrected within 24 hours. The probability of accidents previously evaluated are independent of the OTΔT and OPΔT setpoints. ITS 3.2.2 Required Action A.1.2.1 to reduce THERMAL POWER to

below 50% will likely result in an enthalpy rise hot channel factor that is well below the limiting value at this power level. A reduction of the OPΔT and OTΔT setpoints as required in the Current Technical Specifications (CTS) could potentially result in an OPΔT or OTΔT trip prior to a high neutron flux trip at 55% RTP in response to a transient. While an earlier OPΔT or OTΔT trip may result in a slight improvement in results for the fuel in the accident analyses with respect to a high neutron flux trip, the required reduction in THERMAL POWER provides reasonable assurance that the $F_{\Delta H}$ is maintained within analysis assumptions. Therefore, this change does not involve an increase in the consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   This change eliminates a requirement to reduce the OTΔT and OPΔT setpoints by the fraction $F_{\Delta Hlimit}/F_{\Delta Hactual}$ if the out of limit condition for $F_{\Delta H}$ is not corrected within 24 hours. ITS 3.2.2 Required Action A.1.2.1 to reduce THERMAL POWER to below 50% will likely result in an enthalpy rise hot channel factor that is well below the limiting value at this power level. Further reduction of the OPΔT and OTΔT setpoints is a small contribution to the safety margin, i.e., an OPΔT or OTΔT trip could potentially occur at the reduced setpoint prior to a high neutron flux trip at 55% RTP in response to a transient. While an earlier OPΔT or OTΔT trip could result in a slight improvement in safety margin, this contribution to the safety margin is expected to be small. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

L6 Change

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. This change eliminates the requirement to reduce the high neutron flux setpoint to no greater than 55% of rated power in

the event that the cumulative penalty time for AXIAL FLUX DIFFERENCE (AFD) outside the target band exceeds one hour. The probability of occurrence of an accident is independent of value of the high neutron flux setpoint. Because the ITS retains the requirement to reduce THERMAL POWER under these conditions, the variance in axial peaking factor and axial xenon distribution under these conditions will be minimized. Reduction of the high neutron flux setpoints will not affect the axial peaking factor or axial xenon distribution. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The safety margins relating to AFD are preserved by the combination of Required Actions A.1, B.1, C.1, and D.1, which provide reasonable assurance that unanalyzed xenon axial distributions will not result from a different pattern of xenon buildup and decay, and that the peaking factors assumed in accident analysis will not be exceeded with the existing xenon condition, or otherwise, that THERMAL POWER will be reduced sufficiently to preclude the axial xenon distribution from becoming significantly skewed. Reduction of the high neutron flux setpoint in addition to reducing THERMAL POWER adds no additional safety margin to the core thermal limits because the lower high neutron flux setpoints will not affect the axial peaking factor, axial xenon distribution or analysis assumptions with respect to axial power distribution. Therefore, this change to eliminate the reduction in the high neutron flux setpoints under the stated conditions does not involve a reduction in a margin of safety.

L7 Change

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The proposed change allows up to 16 hours to be

accumulated with AFD outside of the target band without penalty deviation time while the excore detectors are being calibrated. Some deviation from the target band is necessary to perform the calibration of the excore detectors. The AFD is a plant parameter and is not an initiator of any analyzed event. The duration that AFD will be outside the target band for the purposes of excore detector calibration is sufficiently low to provide reasonable assurance that the overall effect on the axial peaking factor and axial xenon distribution will not impact the consequences of analyzed accidents. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The change introduces no new mode of plant operation or changes in the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed 16 hour allowance for AFD to be outside of the target band for the purposes of excore detector calibration is based on operating experience. Some degree of deviation of AFD from the target band is needed for excore detector calibration. The frequency of calibration of the excore detectors is once per 92 days. The process of incore excore calibration involves inducing short term axial power deviations in alternating directions in order to develop a plot of excore detector response. Since the axial power deviations are alternated, the overall effect on axial xenon distribution is small. As such, any reduction in the margin of safety will be insignificant and offset by the benefit of avoiding an unnecessary change in THERMAL POWER during the calibration process. Therefore, this change does not involve a significant reduction in a margin of safety.

L8 Change

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing

normal plant operation. The proposed change relaxes the increased Surveillance Frequency from ½ hour to one (1) hour when the AFD monitor has been out of service for more than 24 hours. Manual logging of the AFD plant parameter is not an initiator of any analyzed event. The one (1) hour Frequency for manual logging of AFD after the first 24 hours is requested in conjunction with the new more restrictive requirement to monitor AFD every 15 minutes for the first 24 hours. Hence, 96 data points will be logged the first 24 hours to assure that no unusual trends in AFD are occurring, before the one hour Frequency begins. After 24 hours, the Surveillance Frequency is reduced to 1 hour because the AFD may deviate from the target band for up to 1 hour using the methodology of Parts B and C of LCO 3.2.3 to calculate the cumulative penalty deviation time before corrective action is required. Since the cumulative penalty time is included in the analyses, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The change involves only a Surveillance Frequency and introduces no new mode of plant operation or changes in the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed one (1) hour Frequency for manually logging AFD after the first 24 hours is based upon the availability of 15 minute data for the first 24 hours to detect any adverse trends in progress that may have been occurring when the AFD monitor became out of service. After 24 hours, the Surveillance Frequency is reduced to 1 hour because the AFD may deviate from the target band for up to 1 hour using the methodology of Parts B and C of LCO 3.2.3 to calculate the cumulative penalty deviation time before corrective action is required. Since the cumulative penalty time is included in the analyses, this change does not involve a reduction in the margin of safety.

L9 Change

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. This change eliminates a Surveillance Frequency for establishing the target AFD when THERMAL POWER is changed by 10% from the level at which F_Q was last measured. The probability of accidents previously evaluated are independent of Surveillance Frequency for the target AFD. The target AFD is sensitive to fuel burnup and not to changes in THERMAL POWER. Because the target AFD is a slowly changing parameter, the ITS Surveillance Frequency of once after each refueling prior to THERMAL POWER exceeding 75% RTP and 31 EFPD thereafter is sufficient to provide reasonable assurance that the target AFD will be accurate for measurement of AFD and that AFD can remain within the assumed values in the accident analyses. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The change is a reduction in Surveillance Frequency. The frequency of surveillance does not create the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   This change eliminates a Surveillance Frequency for the target AFD when THERMAL POWER is changed by 10% from the level at which F_Q was last measured. The target AFD is sensitive to fuel burnup and is a slowly changing parameter. The ITS Surveillance Frequency of once after each refueling prior to THERMAL POWER exceeding 75% RTP and 31 EFPD thereafter is sufficient to assure that changes in the target AFD will not significantly affect the margins from AFD to the target AFD. Therefore, this change does not involve a significant reduction in a margin of safety.

<u>L10 Change</u>

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The proposed change excludes LCO 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," applicability from THERMAL POWER exactly at 50% RTP. The QPTR is a plant parameter and is not an initiator of any analyzed event. The proposed change affects applicability of the LCO at exactly 50% power, and since there is no practical difference between THERMAL POWER at exactly 50% RTP and THERMAL POWER at slightly above 50% RTP, the consequences of an accident with QPTR in excess of 1.02 are not affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The change introduces no new mode of plant operation or changes in the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed restated applicability for QPTR to exclude exactly 50% RTP does not practically differ from an applicability that includes exactly 50% RTP. Similarly, the margin of safety for QPTR at 50% is insignificantly different from the margin of safety for QPTR at THERMAL POWER slightly above 50% RTP. Therefore, this change does not involve a significant reduction in a margin of safety.

L11 Change

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The proposed change eliminates the requirement that the power range high flux setpoint be reduced by 2% for every one (1) percent that QPTR exceeds 1.00. QPTR and high flux setpoints are not

initiators of any analyzed event. The addition of ITS 3.2.4 Required Actions A.2, A.3, A.4, A.5, and A.6, result in continued operation of the plant consistent with the safety analyses. Since the reduced THERMAL POWER requirements maintain the plant consistent with the assumptions in the safety analyses, reduction of the power range high flux setpoints does not adversely affect accident consequences. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The safety margins relating to QPTR are preserved by reducing THERMAL POWER in the condition that QPTR is in excess of 1.02. When the QPTR exceeds its limit, the change in the power distribution can affect rod bank worths and peaking factors for rod malfunction accidents. The remaining Required Actions provides reasonable assurance that plant operations are maintained consistent with the assumptions in the affected safety analyses. Therefore, this change does not involve a significant reduction in a margin of safety.

L12 Change

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The proposed change allows continued operation above 50% RTP with QPTR in excess of the limit as long as the required THERMAL POWER reductions are maintained, the F_Q and $F_{\Delta H}$ limits are maintained, and the QPTR condition remains analyzed for the duration of the cycle. The proposed change also eliminates the requirement to reset the power range high flux setpoints to 55% power. QPTR and high flux setpoints are not initiators of any analyzed event. The addition of ITS 3.2.4 Required Actions A.2, A.3, A.4, A.5, and A.6, result in continued operation of the plant consistent with the safety analyses. Since the

reduced THERMAL POWER requirements maintain the plant consistent with the assumptions in the safety analyses, reducing THERMAL POWER further to below 50% RTP and reducing the power range high flux setpoints does not adversely affect accident consequences. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The change introduces no new mode of plant operation or changes in the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change eliminates further THERMAL POWER reductions and reduction in the power range high flux setpoints when QPTR is in excess of 1.02. When the QPTR exceeds its limit, the change in the power distribution can affect rod bank worths and peaking factors for rod malfunction accidents. The remaining Required Actions provides reasonable assurance that plant operations are maintained consistent with the assumptions in the affected safety analyses. Therefore, this change does not involve a significant reduction in a margin of safety.

L13 Change

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The proposed change eliminates the provision for continued operation with QPTR in excess of 1.09 coincident with a misaligned rod, and eliminates the shutdown requirement if QPTR is in excess of 1.09 without a coincident misaligned rod. QPTR is not an initiator of any analyzed event. Rod group alignment limits are adequately protected by LCO 3.1.4, "Rod Group Alignment Limits." The addition of ITS 3.2.4 Required Actions A.2, A.3, A.4, A.5, and A.6, result in continued operation of the plant consistent with the safety analyses. Since the reduced THERMAL POWER requirements maintain the plant consistent

with the assumptions in the safety analyses, reducing THERMAL POWER further by 2% RTP for every percent that QPTR exceeds 1.09 or entering MODE 3 does not adversely affect accident consequences. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The change introduces no new mode of plant operation or changes in the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change eliminates the provision for continued operation with QPTR in excess of 1.09 coincident with a misaligned rod, and eliminates the shutdown requirement if QPTR is in excess of 1.09 without a coincident misaligned rod. When the QPTR exceeds its limit, the change in the power distribution can affect rod bank worths and peaking factors for rod malfunction accidents. When QPTR is in excess of 1.09, these effects are more severe: however, the Required Actions of LCO 3.2.4 provides reasonable assurance that plant operations are maintained consistent within bounds of the safety analyses. Therefore, this change does not involve a significant reduction in a margin of safety.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
NRC DOCKET NO. 50-261/LICENSE NO. DPR-23
REQUEST FOR TECHNICAL SPECIFICATION CHANGE
CONVERSION TO IMPROVED STANDARD TECHNICAL SPECIFICATIONS

ENVIRONMENTAL CONSIDERATIONS

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulator actions for categorical exclusion for performing an environmental assessment. A proposed change for an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; (3) result in an increase in individual or cumulative occupational radiation exposure. We have reviewed this request and determined that the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance with the amendment. The basis for this determination follows.

Proposed Change

This request proposes to change the technical specifications to be consistent with NUREG-1431; Standard Technical Specifications, Westinghouse Plants Revision 1, 04/07/95 within limitations imposed by plant specific design and licensing basis.

Basis

The proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons.

1. As demonstrated in the No Significant Hazards Evaluation, the proposed changes do not involve a significant hazards consideration.

2. These proposed changes are being made to establish consistency with the improved Standard Technical Specifications (ISTS) - Westinghouse Plants, NUREG 1431, Rev. 1, including approved generic changes, and do not involve physical changes to the facility, nor do they affect actual plant effluents.

3. These proposed changes are being made to establish consistency with the improved Standard Technical Specifications (ISTS) - Westinghouse Plants, NUREG 1431, Rev. 1, including approved generic changes and do not involve physical changes to the facility, and they do not significantly affect individual or cumulative occupational radiation exposures.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.2 - POWER DISTRIBUTION LIMITS

PART 4

MARKUP OF NUREG 1431, REVISION 1, "STANDARD TECHNICAL SPECIFICATIONS - WESTINGHOUSE PLANTS" (ISTS)

3.2 POWER DISTRIBUTION LIMITS

3.2.1A Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_{xy} Methodology)

LCO 3.2.1A $F_Q(Z)$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. $F_Q(Z)$ not within limit.	A.1	Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q(Z)$ exceeds limit.	15 minutes
	AND		
	A.2	Reduce AFD acceptable operation limits by the percentage $F_Q(Z)$ exceeds limit.	4 hours
	AND		
	A.3	Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q(Z)$ exceeds limit.	8 hours
	AND		
	A.4	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q(Z)$ exceeds limit.	72 hours
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.5 Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify measured values of $F_Q(Z)$ are within limits.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	----------NOTES---------- 1. If $F^C_{xy} > F^L_{xy}$, evaluate the effect of F_{xy} on the predicted F_Q^{PQ} to determine if $F_Q(Z)$ is within its limits. 2. If $F^{RTP}_{xy} < F^C_{xy} \leq F^L_{xy}$, SR 3.2.1.2 shall be repeated within 24 hours after an increase in THERMAL POWER at which F^C_{xy} was last determined, of at least 20% RTP. ---------- Verify $F^C_{xy} < F^L_{xy}$.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND 31 EFPD thereafter

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.1Q Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)

LCO 3.2.1Q [3.10.2.1] $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limits specified in the COLR. $F_Q^V(Z)$

[3.10.2.1] APPLICABILITY: MODE 1.

[3.10.2.2.1.a] ACTIONS

A.1 Reduce AFD target band limits to restore $F_Q^V(Z)$ to within limit OR

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. $F_Q(Z)$ not within limit. [3.10.2.2.1.b]	A.1 (2.1) Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q(Z)$ exceeds limit.	15 minutes 30
	AND	
[M2]	A.2 (2.2) Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q(Z)$ exceeds limit.	8 hours
	AND	
[3.10.2.1.1]	A.3 (2.3) Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q(Z)$ exceeds limit.	72 hours
	AND	
[M2]	A.4 (2.4) Perform SR 3.2.1.1.	Prior to increasing THERMAL POWER above the limit of Required Action A.1 (A.2.1)

(continued)

CTS

2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. $F_Q^W(Z)$ not within limits.	B.1 Reduce AFD limits ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	2 hours 4
[M2] A Required Action and associated Completion Time not met.	A.1 Be in MODE 2. B	6 hours

2

SURVEILLANCE REQUIREMENTS

[3.10.2.1.1] ----------NOTE----------
During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE	FREQUENCY
~~SR 3.2.1.1 Verify $F_Q^C(Z)$ is within limit.~~	~~Once after each refueling prior to THERMAL POWER exceeding 75% RTP~~ ~~AND~~ ~~Once within [12] hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified~~ ~~AND~~ ~~31 EFPD thereafter~~

3

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 ~~NOTE~~ ~~If $F_Q^W(Z)$ is within limits and measurements indicate~~ ~~maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)}\right]$~~ ~~has increased since the previous evaluation of $F_Q^C(Z)$:~~ ~~a. Increase $F_Q^W(Z)$ by a factor of [1.02] and reverify $F_Q^W(Z)$ is within limits; or~~ ~~b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive flux maps indicate~~ ~~maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)}\right]$~~ ~~has not increased.~~ [3.10.2.1.1] Verify $F_Q^W(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND (continued)

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued)	Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q(Z)$ was last verified AND 31 EFPD thereafter

CTS

2

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

[3.10.2.1]

[3.10.2.1] APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ---------- [M11] $F^N_{\Delta H}$ not within limit. [M11] [3.10.2.1.1]	A.1.1 Restore $F^N_{\Delta H}$ to within limit.	4 hours
	OR	
	A.1.2.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND	
	A.1.2.2 Reduce Power Range Neutron Flux–High trip setpoints to ≤ 55% RTP.	8 hours
	AND	
[M11]	A.2 Perform SR 3.2.2.1.	24 hours
	AND	
		(continued)

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued) [M11]	A.3 --------NOTE--------- THERMAL POWER does not have to be reduced to comply with this Required Action. ---------- Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP <u>AND</u> Prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 24 hours after THERMAL POWER reaching ≥ 95% RTP
[M11] B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

CTS

2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1 [3.10.2.1.1]	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND 31 EFPD thereafter

INSERT I3.2.2-1

7

[3.10.2.2.3]

INSERT I3.2.2-1

----------NOTE----------

If $F^N_{\Delta H}$ is within limits and measurements indicate that $F^N_{\Delta H}$ is increasing with exposure then:

a. Increase $F^V_Q(Z)$ by a factor of 1.02 and reverify $F^V_Q(Z)$ is within limits; or

b. Perform SR 3.2.1.1 and SR 3.2.3.3 once per 7 EFPDs until two successive measurements indicate $F^N_{\Delta H}$ is not increasing.

2

3.2 POWER DISTRIBUTION LIMITS

PDC-3 Axial Offset Control Methodology — 8

3.2.3A AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)

LCO 3.2.3 The AFD:

[3.10.2.5] [3.10.2.11]

a. Shall be maintained within the target band about the target flux difference. The target band is specified in the COLR.

[3.10.2.4]

----------NOTE----------
The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.

[3.10.2.7 and 3.10.2.7.a]

b. May deviate outside the target band with THERMAL POWER < 90% RTP (or 0.9 APL, whichever is less) but ≥ 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR. — 9

10

[A13] [3.10.2.2.2]

----------NOTE (5)----------
1. Penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

INSERT 3.2.3-1

[3.10.2.8.a]

c. May deviate outside the target band with THERMAL POWER < 50% RTP.

[3.10.2.8.b]

----------NOTE----------
Penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

[3.10.2.5]

APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP.

[3.10.2.6] [3.10.2.7]

----------NOTE----------
A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.6, provided AFD is maintained within acceptable operation limits.

[3.60.2.9]

ITS INSERT 3.2.3-1

2. The Allowable Power Level (APL) is the limitation placed on THERMAL POWER for the purposes of applying the AFD target flux and operational limit curves. The APL is as follows:

$$APL = \text{minimum over } Z \text{ of } (100\%)(F_Q^{RTP}(Z))(K(Z))/F_Q^{V}(Z)$$

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ 90% RTP or Ø.9 APL, whichever is less [3.10.2.6] AND AFD not within the target band.	A.1 Restore AFD to within target band.	15 minutes 9
[3.10.2.6] B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 90% RTP. or Ø.9 APL, whichever is less	15 minutes 9
[A14] C. ---------NOTE--------- Required Action C.1 and C.2 must be completed whenever Condition C is entered. [3.10.2.7.a] [3.10.2.8.b] ---------- THERMAL POWER < 90% and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours. [3.10.2.7.a] OR THERMAL POWER < 90% and ≥ 50% RTP with AFD not within the acceptable operation limits. [M15]	C.1 Reduce THERMAL POWER to < 50% RTP. 16 AND C.2 Restore cumulative penalty deviation time to less than 1 hour RTP or Ø.9 APL, whichever is less	30 minutes 16 Prior to increasing THERMAL POWER to ≥ 50% RTP 9

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. ---------NOTE--------- Required Action D.1 must be completed whenever Condition D is entered. ---------- Required Action and associated Completion Time for Condition C not met.	D.1 Reduce THERMAL POWER to < 15% RTP.	9 hours

[M18]

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD is within limits for each OPERABLE excore channel.	7 days

[M18]

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.3.2 [3.10.2.10]	----------NOTE---------- Assume logged values of AFD exist during the preceding time interval. ----------	
[3.10.2.10]	Verify AFD is within limits and log AFD for each OPERABLE excore channel.	-----NOTE----- Only required to be performed if AFD monitor alarm is inoperable ----------
[M20]		Once within 15 minutes and every 15 minutes thereafter when THERMAL POWER ≥ 90% RTP
		AND
[L8]		Once within 1 hour and every 1 hour thereafter when THERMAL POWER < 90% RTP or 0.9 APL, whichever is less
~~SR 3.2.3.3~~	~~Update target flux difference of each OPERABLE excore channel by:~~ ~~a. Determining the target flux difference in accordance with SR 3.2.3.4, or~~ ~~b. Using linear interpolation between the most recently measured value, and either the predicted value for the end of cycle or 0% AFD.~~	~~Once within 31 EFPD after each refueling~~ ~~AND~~ ~~31 EFPD thereafter~~

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.3.4 3 [3.10.2.3]	----------NOTE---------- 1. The initial target flux difference after each refueling may be determined from design predictions. ---------- 11	
[3.10.2.1.1]	Determine, by measurement, the target flux difference of each OPERABLE excore channel.	Once within 31 EFPD after each refueling AND 92 31 EFPD thereafter 11

[3.10.2.3] 2. The target flux difference shall be determined in conjunction with the measurement of $F_Q(Z)$ in accordance with SR 3.2.1.1 12

3.2 POWER DISTRIBUTION LIMITS

3.2.3B AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits specified in the COLR.

NOTE

The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	7 days AND Once within 1 hour and every 1 hour thereafter with the AFD monitor alarm inoperable

2

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

[M26] LCO 3.2.4 The QPTR shall be ≤ 1.02.

[3.10.3.1] APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. QPTR not within limit. [3.10.3.1a]	A.1	Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours
	AND		
[M26]	A.2	Perform SR 3.2.4.1 and reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	Once per 12 hours
	AND		
[M26]	A.3	Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours AND Once per 7 days thereafter
	AND		
[M26]	A.4	Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND		

(continued)

2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued) [M26]	A.5 --------NOTE--------- Perform Required Action A.5 only after Required Action A.4 is completed. ----------	
[M26]	Calibrate excore detectors to show zero QPTR. Normalize excore detectors to show zero QPTR	Prior to increasing THERMAL POWER above the limit of Required Action A.1 or A.2 (13) (14)
	AND	
[M26]	A.6 --------NOTE--------- Perform Required Action A.6 only after Required Action A.5 is completed. ----------	
[M26]	Perform SR 3.2.1.1 and SR 3.2.2.1.	Within 24 hours after reaching RTP OR Within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1 or A.2 (14)
B. Required Action and associated Completion Time not met. [3.10.3.1.6]	B.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

CTS

2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1 [1.8] [M26] [M26]	----------NOTES---------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. ~~If adequate Power Range Neutron Flux channel inputs are not OPERABLE.~~ 15 ----------	
	Verify QPTR is within limit by calculation.	7 days AND Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable
SR 3.2.4.2 [M26]	----------NOTE---------- Only required to be performed if input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER ≥ 75% RTP. ----------	
[M26]	Verify QPTR is within limit using the movable incore detectors.	Once within 12 hours AND 12 hours thereafter

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.2 - POWER DISTRIBUTION LIMITS

PART 5

JUSTIFICATION FOR DIFFERENCES (JFDs) TO ISTS

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain sections of NUREG-1431, "Improved Standardized Technical Specifications - Westinghouse Plants," Revision 1, (ISTS) are not incorporated into ITS because they are not applicable. Specifically, ISTS Section 3.2.1A, "Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_{xy} Methodology), " and Section 3.2.3B, "Axial Flux Difference (AFD) (Relaxed Axial Offset Control (RAOC) Methodology), are not applicable to the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No 2, and are not included in the ITS.

2 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences and conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes which involve the insertion of plant specific terms or parameters are used to preserve consistency with the CTS and licensing basis.

3 The expression F_Q^W is replaced with $F_Q^V(Z)$; the expression F_Q^C is not used; and, ISTS Surveillance Requirement (SR) 3.2.1.1 is not included in the ITS, to be consistent with the PDC-3 axial offset control methodology. The expression $F_Q^V(Z)$ is equivalent to F_Q^W utilized in the ISTS for Constant Axial Offset Control (CAOC) Methodology. The expression F_Q^C as used in the CAOC methodology is bounded by the other expressions of $F_Q(Z)$ in the PDC-3 axial offset control methodology. ISTS SR 3.2.1.2 is included in ITS as SR 3.2.1.1, which verifies that $F_Q^V(Z)$ is within the limits to satisfy Limiting Condition for Operations (LCO) 3.2.1.

4 ITS 3.2.1 Required Action A.1 is added to reduce AFD target band limits to restore F^V_Q (Z) to within limits and to delete Required Action B.1 as presented in the ISTS. The PDC-3 axial offset control methodology provides two distinct target bands for operation which consist of a ± 3% target band and a ± 5% target band. Required Action B.1 as presented in ISTS is invalid and is not used in the ITS, as there is no allowance in the PDC-3 axial offset control methodology to interpolate between the two allowable target bands.

5 ISTS Specification 3.2.1 is modified to increase the Completion Time for Required Action A.2.1 to 30 minutes to allow for accomplishment of ITS 3.2.1 Required Action A.1 before Required Actions A.2.1 through A.2.4. In the event that the out of limit condition cannot be corrected within 15 minutes by operating at the more restrictive target band then the remaining Required Actions in Condition A to reduce THERMAL POWER and reduce ΔT trip setpoints are required.

6 ISTS Specification 3.2.1 is revised to include the requirement to additionally reduce the Overtemperature Delta-Temperature (OTΔT) setpoint in ITS 3.2.1 Required Action A.2.3, in order to maintain consistency with the safety analyses and the Current Licensing Basis.

7 ISTS Specification 3.2.1 is modified to delete an applicability Note that relates to trending $F_Q^C(Z)/K(Z)$, and ISTS Specification 3.2.2 is modified to include an applicability Note to SR 3.2.2.1 to account for trending of $F_{\Delta H}^N$. The PDC-3 axial offset control methodology utilizes $F_{\Delta H}^N$ to trend the approach to power distribution limits rather than $F_Q^C(Z)/K(Z)$, which is used in the CAOC methodology.

8 ISTS Specification 3.2.3 is modified in title to be "Axial Flux Difference (AFD) (PDC-3 Axial Offset Control Methodology)," to replace the term "Constant Axial Offset Control (CAOC)." The PDC-3 axial offset control methodology is the approved methodology for neutronic calculations for Siemens Power Corporation manufactured fuel for HBRSEP. CAOC refers to the approved methodology for Westinghouse manufactured fuel.

9 ISTS Specification 3.2.3 is modified throughout where references to 90% RATED THERMAL POWER (RTP) occur to also require that THERMAL POWER be less than 0.9 Allowable Power Level (APL), whichever is less. The limitation to 0.9 APL is necessary to keep Axial Flux Difference (AFD) deviations from the target band within the requirements of the PDC-3 axial offset control methodology.

10 ISTS Specification 3.2.3 is modified in a note to LCO 3.2.3 to include the APL limitation that applies to THERMAL POWER when determining the AFD to its target band. The PDC-3 axial offset control methodology requires that the AFD limitation curves be adjusted when the APL is less than 100%.

11 ISTS SR 3.2.3.3 is deleted, and subsequent SR are renumbered. The Specification is also modified by changing the Frequency of ITS SR 3.2.3.3 to 31 Effective Full Power Days (EFPDs) from 92 EFPDs. The PDC-3 axial offset control methodology does not allow the use of linear interpolation to determine the target flux values. The Frequency is increased in order to provide an adequate interval between measurement of the target flux difference in the absence of an approved interpolative method.

12 ISTS Specification 3.2.3 is modified by adding Note 2 to ITS SR 3.2.3.3 to require that the target flux difference be determined in conjunction with the measurement of the heat flux hot channel factor, $F_Q(Z)$, in accordance with ITS SR 3.2.1.1. The performance of SR 3.2.3.3. in conjunction with SR 3.2.1.1 is a requirement of the PDC-3 axial offset control methodology.

13 ISTS Specification 3.2.4 is modified to replace the term, "calibrated excore detectors to show a zero tilt," with, "normalize excore detectors to eliminate the tilt," in order to clarify that the measured QUADRANT POWER TILT RATIO (QPTR) need not precisely equate to zero prior to increasing THERMAL POWER above the level determined by ITS 3.2.4 Required Action A.1, and that the Required Action is a normalization of excore detector indications rather than a calibration, i.e., performance of SR 3.3.1.10.

14 ISTS Specification 3.2.4 is modified to include applicability of ITS 3.2.4 Required Action A.2 to the Completion Time for Required Actions A.5 and A.6, to reflect that THERMAL POWER limitations from either Required Action A.1 or A.2 may be more limiting.

15 ISTS Specification 3.2.4 is modified in Note 2 to ITS SR 3.2.4.1 to allow the performance of ITS SR 3.2.4.2 in lieu of ITS SR 3.2.4.1 at any time. Since the verification of QPTR is most accurately performed with the incore detector system, performance of SR 3.2.4.2 is the preferred method to verify QPTR.

16 ISTS Specification 3.2.3 is modified to include Required Action C.2 to ensure consistency in the analyses performed in accordance with the PDC-3 axial offset control methodology.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.2 - POWER DISTRIBUTION LIMITS

PART 6

MARKUP OF ISTS BASES

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1A Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_{xy} Methodology)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height of the core (Z).

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions adjusted for uncertainty. Therefore, $F_Q(Z)$ is a measure of the peak pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. Therefore, these LCOs preserve core limits on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system, and measurements are generally taken with the core at or near steady state conditions.

With the measured three dimensional power distributions, it is possible to determine a measured value for $F_Q(Z)$. However, because this value represents a steady state condition, it does not include variations in the value of $F_Q(Z)$, which are present during a nonequilibrium situation such as load following.

The steady state value of the fundamental radial peaking factor (F_{xy}) is adjusted by an elevation dependent factor to account for the variations in $F_Q(Z)$ due to transient conditions.

Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the total peaking factor assumed as an initial condition in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ assumed in safety analyses for other accidents. Therefore, this LCO provides conservative limits for other accidents.

$F_Q(Z)$ satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{CFQ}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{CFQ}{0.5} K(Z) \qquad \text{for } P \leq 0.5$$

where: CFQ is the F_Q limit at RTP provided in the COLR.

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

For this facility, the actual values of CFQ and K(Z) are given in the COLR; however, CFQ is normally a number on the order of [2.32], and K(Z) is a function that looks like the one provided in Figure B 3.2.1A-1.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ may produce unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained while in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is insufficient stored energy in the fuel or energy being transferred to the

(continued)

BASES

APPLICABILITY (continued)

reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% for each 1% by which $F_Q(Z)$ exceeds its limit maintains an acceptable absolute power density. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

A.2

When core peaking factors are sufficiently high that LCO 3.2.3 does not permit operation at RTP, the Acceptable Operation Limits for AFD are scaled down. This percentage reduction is equal to the amount, expressed as a percentage, by which $F_Q(Z)$ exceeds its specified limit. This ensures a near constant maximum linear heat rate in units of kilowatts per foot at the acceptable operation limits. The Completion Time of 4 hours for the change in setpoints is sufficient, considering the small likelihood of a severe transient in this relatively short time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.3

A reduction of the Power Range Neutron—High trip setpoints by ≥ 1% for each 1% by which $F_Q(Z)$ exceeds its specified limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 8 hours is sufficient, considering the small likelihood of a severe transient in this period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

(continued)

BASES

ACTIONS (continued)

A.4

Reduction in the Overpower ΔT trip setpoints by ≥ 1% for each 1% by which $F_Q(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.5

Verification that $F_Q(Z)$ has been restored to within its limit by performing SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1 ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

B.1

If the Required Actions of A.1 through A.4 cannot be met within their associated Completion Times, the plant must be placed in a MODE or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

Verification that $F_Q(Z)$ is within its limit involves increasing the measured values of $F_Q(Z)$ to allow for manufacturing tolerance and measurement uncertainties and then making a comparison with the limits. These limits are provided in the COLR. Specifically, the measured value of the Heat Flux Hot Channel Factor (F_Q^M) is increased by 3% to account for fuel manufacturing tolerances and by 5% for flux

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

map measurement uncertainty. This procedure is equivalent to increasing the directly measured values of $F_Q(Z)$ by 1.0815% before comparing with LCO limits (Ref. 4).

Performing the Surveillance in MODE 1 prior to THERMAL POWER exceeding 75% RTP after each refueling ensures that $F_Q(Z)$ is within limit when RTP is achieved.

The Frequency of 31 EFPD is adequate for monitoring the change of power distribution with core burnup because the power distribution changes relatively slowly for this amount of fuel burnup. The Surveillance may be done more frequently if required by the results of SR 3.2.1.2.

SR 3.2.1.2

The nuclear design includes calculations that predict that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken at steady state conditions, the axial variations in power distribution for normal operation maneuvers such as load following are not present in the flux map data. These axial variations are, however, conservatively calculated by considering, in the nuclear design process, a wide range of unit maneuvers in normal operation. $F_{xy}(Z)$ is the radial peaking factor, which is one component of $F_Q(Z)$ and should be consistent between the nuclear design values and the measured values. ($F_{xy}(Z)$ multiplied by the normalized average axial power at elevation Z gives $F_Q(Z)$.)

The core plane regions applicable to an F_{xy} evaluation exclude the following, measured in percent of core height:

a. Lower core region, from 0% to 15% inclusive;

b. Upper core region, from 85% to 100% inclusive;

c. Grid plane regions, ± 2% inclusive; and

d. Core plane regions, within ± 2% of the bank demand position of the control banks.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

The following terms are used in the F_{xy} evaluation:

F^M_{xy} = The measured value of F_{xy} obtained directly from the flux map results.

F^C_{xy} = The measured value, F^M_{xy}, multiplied by 1.0815 to account for fuel manufacturing tolerances and flux map measurement uncertainty (Ref. 2).

F^{RTP}_{xy} = The limit of F_{xy} at RTP.

F^L_{xy} = F^{RTP}_{xy}[(1 + PFXY)(1 - P)] (the limit of F_{xy} at the current THERMAL POWER level).

PFXY = The power factor multiplier for F_{xy}.

P = [The Fraction of RTP at which F_{xy} was measured.]

F^{PR}_Q = The predicted value of the Heat Flux Hot Channel Factor.

F^{RTP}_{xy} and PFXY are provided in the COLR. F^M_{xy} and F^C_{xy} are measured and calculated at discrete core elevations. Note that F_{xy} can be rewritten as $F_{xy}(Z)$ to indicate that F_{xy} varies along the axial height of the core. Flux map data are typically taken for 30 to 75 core elevations.

The top and bottom regions of the core are excluded from the F_{xy} evaluation because of the difficulty of making precise and meaningful measurements in these regions and also because of the low probability that these regions would be more limiting than the central 70% of the core in the accident analyses.

Grid plane regions and rod tip regions are also excluded because the flux data may give spurious values because of the difficulty in lining up flux traces accurately in regions of rapidly varying flux. In addition, these small portions of the core are reduced in local power density because of neutron absorption in the grids and control rods and, therefore, cannot be regions of peak linear power.

An evaluation of $F_{xy}(Z)$ is used to confirm that $F_Q(Z)$ is within its limits. If F^C_{xy} is < F^{RTP}_{xy}, it is concluded that

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

the LCO limit on $F_Q(Z)$ is met. This result is true for flux maps taken at reduced power because the $F_{xy}(Z)$ value is inherently decreased as THERMAL POWER is increased. The feedback from the Doppler coefficient and moderator effects flattens the power distribution with increased THERMAL POWER.

The first Note of this Surveillance provides the action to be taken if F^C_{xy} is > F^L_{xy}. In this case, the $F_Q(Z)$ limit may be exceeded. Proportionally increasing the predicted $F^{PR}_Q(Z)$ by the amount that F^L_{xy} is exceeded gives an adjusted $F_Q(Z)$, which is compared with the $F_Q(Z)$ limit. If the adjusted $F_Q(Z)$ exceeds the LCO limit, the operator must perform Required Actions A.1 through A.5.

The second Note in this Surveillance states that if F^C_{xy} is > F^{RTP}_{xy} but < F^L_{xy}, then this Surveillance shall be repeated within 24 hours after exceeding by ≥ 20% RTP the THERMAL POWER at which F^C_{xy} was last determined, so as to demonstrate that $F_{xy}(Z)$ is being sufficiently reduced as power increases. This reduction, because of feedback from the Doppler coefficient and moderator effects, ensures that when RTP is attained, the measured $F^M_{xy}(Z)$ is < F^{RTP}_{xy}.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP after each refueling ensures that the $F_Q(Z)$ limit is met when RTP is achieved.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because the power distribution changes relatively slowly for this amount of fuel burnup. The Surveillance may be done more frequently if required by the results of F_{xy} evaluations. Specifically, the F_{xy} evaluation is required by this Surveillance if the evaluation shows that F^{RTP}_{xy} < F^C_{xy} and to demonstrate that the LCO is met after its limit has been exceeded.

REFERENCES

1. 10 CFR 50.46.

2. Regulatory Guide 1.77, Rev. [ ].

(continued)

BASES

REFERENCES (continued)

3. 10 CFR 50.46, GDC 26.

4. [WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.]

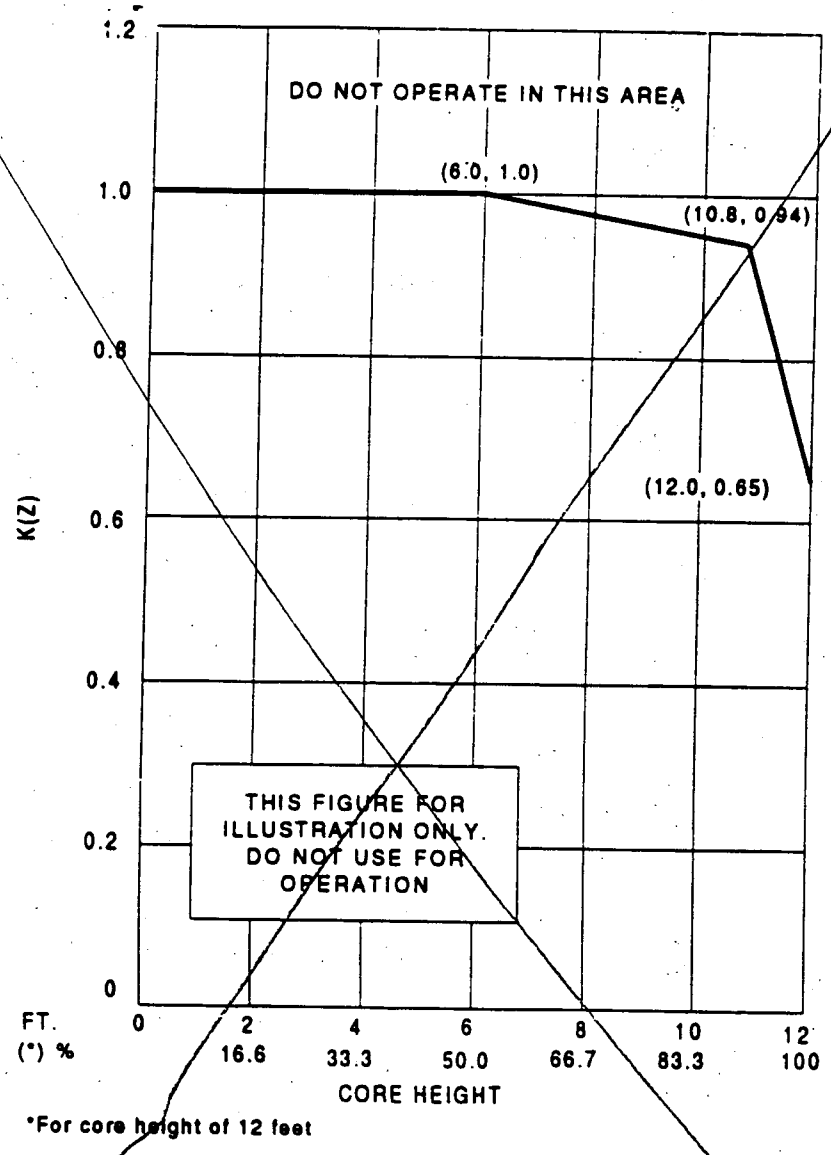


Figure B 3.2.1A-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1~~0~~ Heat Flux Hot Channel Factor ($F_Q(Z)$) (~~F_Q Methodology~~)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," (PDC-3 Axial Offset Control Methodology) and LCO 3.2.4, "QUADRANT ~~POWER~~ TILT POWER RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.~~4~~ 6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(Z)$ that are present during nonequilibrium situations, such as load following.

To account for these possible variations, the steady state value of $F_Q(Z)$ is adjusted by an elevation dependent factor that accounts for the calculated worst case transient conditions.

Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of

(continued)

BACKGROUND (continued)

the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition (Ref. 2)

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and (3)

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3). (4)

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of the NRC Policy Statement.

(continued)

2

BASES (continued)

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \le \frac{CFQ}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \le \frac{CFQ}{0.5} K(Z) \qquad \text{for } P \le 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR.

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

3

For this facility, the actual values of CFQ and K(Z) are given in the COLR; however, CFQ is normally a number on the order of ~~[2.32]~~ 2.5, and K(Z) is a function that looks like the one provided in Figure B 3.2.1B-1.

the PDC-3 constant axial offset control

For ~~Relaxed Axial Offset Control~~ $F_Q^V(z)$ operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and ~~$F_Q^M(Z)$~~. Thus, both $F_Q^C(Z)$ and ~~$F_Q(Z)$~~ must meet the preceding limits on $F_Q(Z)$.

4

$F_Q^V(z)$ is the most limiting expression of $F_Q(z)$. By requiring $F_Q^V(z)$ to be within limits, $F_Q^C(z)$ can be assured to be within limits in steady state and transient situations.

An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value ($F_Q^M(Z)$) of $F_Q(Z)$. Then,

4

$$F_Q^C(Z) = F_Q^M(Z)\ 1.0815$$

where 1.0815 is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

(continued)

BASES

LCO (continued)

The expression for $F_Q^{W}(Z)$ is:

$$F_Q^{W}(Z) = F_Q^{C}(Z)\,W(Z)$$

where $W(Z)$ is a cycle dependent function that accounts for power distribution transients encountered during normal operation. $W(Z)$ is included in the COLR.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

Insert B.3.2.1-1

ACTIONS

A.1 2.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^{C}(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^{C}(Z)$ is $F_Q^{M}(Z)$ multiplied by a factor ~~accounting for manufacturing tolerances and measurement uncertainties~~ and the maneuvering penalty factor V(Z) as stated in the COLR. $F_Q^{M}(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. 30

(continued)

INSERT B.3.2.1-1

A.1

The PDC-3 axial offset control methodology provides two distinct target bands for operation which are the ±3% and the ±5% target bands. The target band that is selected determines the V(Z) penalty to be applied in the calculation of $F_Q^V(Z)$. When operation is restricted to the ±3% target band, the V(Z) penalty is minimized, and the $F_Q^V(Z)$ is reduced. Thus when operation is restricted to the more restrictive target band, the result may be that $F_Q^V(Z)$ is within limits, and no reduction in THERMAL POWER is required. In the event that the reduced target band does not result in an acceptable $F_Q^V(Z)$, the THERMAL POWER will be reduced in accordance with Required Action A.2.1. The Completion Time of 15 minutes provides an acceptable time to reevaluate $F_Q^V(Z)$ within the more restrictive target band to determine if $F_Q^V(Z)$ remains within limits.

ACTIONS (continued)

~~A.2~~ A.2.2

A reduction of the Power Range Neutron Flux–High trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 8 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

~~A.2~~ A.2.3

Reduction in the Overpower and overtemperature ΔT trip setpoints by ≥ 1% for each 1% by which $F_Q^C(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

~~A.4~~ A.2.4

Verification that $F_Q^C(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 prior to increasing THERMAL POWER above the limit imposed by Required Action ~~A.1~~ A.2.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

~~B.1~~

~~If it is found that the maximum calculated value of $F_Q(Z)$ that can occur during normal maneuvers, $F_Q^W(Z)$, exceeds its specified limits, there exists a potential for $F_Q^C(Z)$ to become excessively high if a normal operational transient occurs. Reducing the AFD by ≥ 1% for each 1% by which $F_Q^W(Z)$ exceeds its limit within the allowed Completion Time of 2 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factors are not exceeded.~~

(continued)

BASES

ACTIONS (continued)

~~A~~B.1

If Required Actions ~~A.1 through A.4 or B.1~~ of Condition A are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 ~~and SR 3.2.1.2~~ is ~~are~~ modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that ~~$F_Q(Z)$ and~~ $F_Q^W(Z)$ ~~are~~ is within ~~their~~ specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which ~~they were~~ it was last verified to be within specified limits. Because ~~$F_Q(Z)$ and~~ $F_Q^W(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of ~~$F_Q(Z)$ and~~ $F_Q^W(Z)$ ~~are~~ is made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of ~~$F_Q(Z)$ and~~ $F_Q^W(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of ~~$F_Q(Z)$ and~~ $F_Q^W(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which F_Q was last measured.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.2.1.1~~

~~Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z)$ [1.0815] (Ref. 4). $F_Q^C(Z)$ is then compared to its specified limits.~~

~~The limit with which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.~~

~~Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q^C(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.~~

~~If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required [12] hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).~~ 7

~~The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).~~

SR 3.2.1.~~2~~ 1

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called ~~W~~(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by ~~W~~(Z) gives the

V

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1~~2~~ (continued)

maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit with which $F_Q^W(Z)$ is compared varies inversely with power and directly with the function K(Z) provided in the COLR.

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to ~~15%~~ 10% inclusive; and

b. Upper core region, from ~~85~~ 90 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

~~This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated and found to be within its limit, an evaluation of the expression below is required to account for any increase to $F_Q^W(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.~~

~~If the two most recent $F_Q(Z)$ evaluations show an increase in the expression~~

$$\text{maximum over } Z \left[\frac{F_Q^C(Z)}{K(Z)} \right],$$

~~it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by a factor of [1.02], or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.~~

(continued)

2

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.~~2~~ 1 (continued)

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q(Z)$ limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

$F_Q(Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, ~~12~~24 hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.

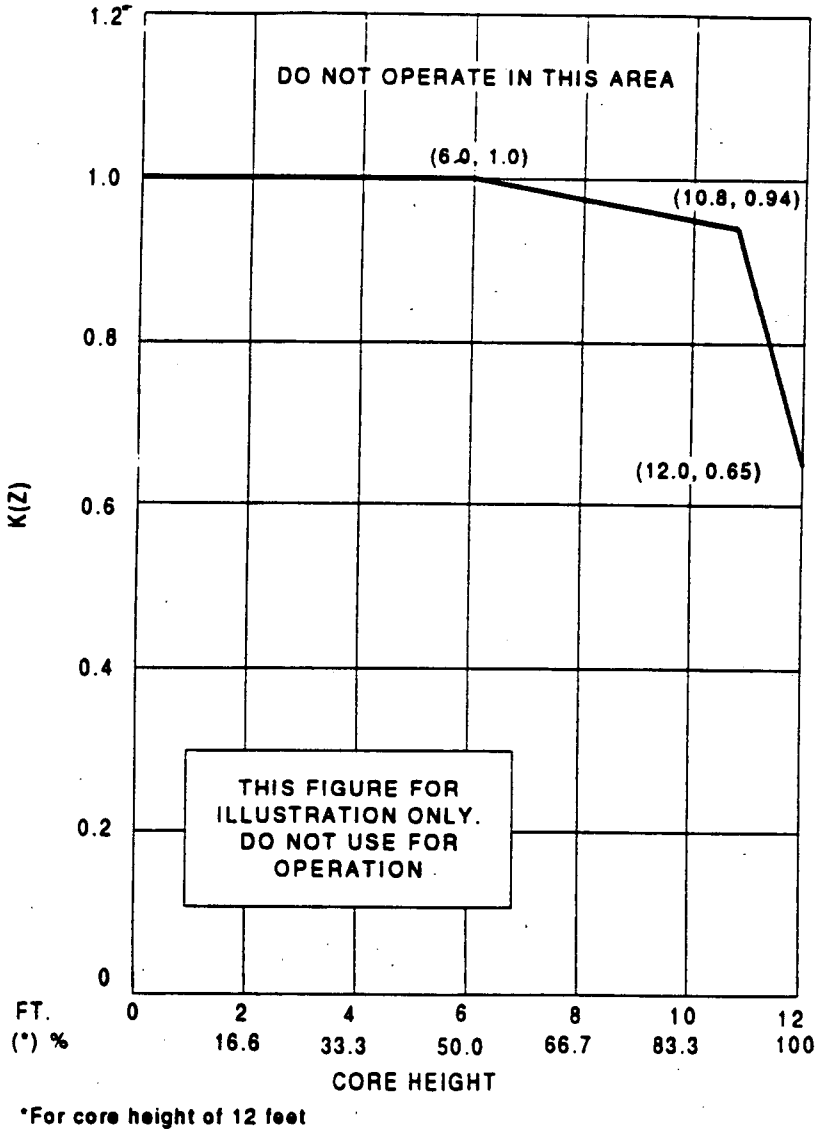
The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46, 1974.
2. UFSAR Section 4.4.2.1
3. ~~2.~~ ~~Regulatory Guide 1.77, Rev. 0, May 1974~~ UFSAR Section 15.4.8
4. ~~3.~~ ~~10 CFR 50, Appendix A, GDC 26~~ UFSAR Section 3.1.
~~4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988~~

2



*For core height of 12 feet

Figure B 3.2.1B-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

2

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to ~~[1.73]~~ 1.154 using the ~~[W3] CHF correlation~~ Advanced Nuclear Fuels Corporation's DNB correlation (i.e., ANFP). All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

(continued)

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition; (Ref. 1)

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 10); and

d. Fuel design limits required by GDC 26 (Ref. 8) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion of 1.31 using the WRB-1 correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use

(continued)

APPLICABLE SAFETY ANALYSES (continued)

this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of the NRC Policy Statement.

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced

(continued)

BASES

LCO (continued)

thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.2% for every 1% RTP reduction in THERMAL POWER. 11

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

ACTIONS

A.1.1

With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time.

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, if power is not reduced because this Required Action is completed within the 4 hour time period, Required Action A.2 nevertheless requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.

However, if power is reduced below 50% RTP, Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding

(continued)

BASES

ACTIONS

A.1.1 (continued)

75% RTP, and within 24 hours after reaching or exceeding 95% RTP. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.

A.1.2.1 and A.1.2.2

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux–High to ≤ 55% RTP in accordance with Required Action A.1.2.2. Reducing RTP to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 8 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

A.2

Once the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which

(continued)

BASES

ACTIONS

A.2 (continued)

is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

Insert B.3.2.2+

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

REFERENCES

1. ~~Regulatory Guide 1.17, Rev. [0], May 1974~~ UFSAR Section 4.4.2.1
2. ~~10 CFR 50, Appendix A, GDC 26~~ UFSAR Section 15.4.8
3. UFSAR Section 3.1
4. 10 CFR 50.46.

INSERT B3.2.2-1

This Surveillance is modified by a Note that may require that the evaluation of $F_Q^V(Z)$ against its limits be performed with a penalty factor or that more frequent surveillances be performed. If $F_{\Delta H}^N$ is within limits and measurements indicate that $F_{\Delta H}^N$ is increasing with exposure, then $F_Q^V(Z)$ is increased by a factor of 1.02, and $F_Q^V(Z)$ is then reverified to be within limits: or, SR 3.2.1.1 and SR 3.2.3.3 are performed once per 7 EFPDs until two successive measurements of $F_{\Delta H}^N$ show that $F_{\Delta H}^N$ is not increasing. These alternative requirements prevent $F_Q^V(Z)$ from exceeding its limit for any significant period of time during the surveillance interval.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3~~B~~ AXIAL FLUX DIFFERENCE (AFD) (~~Constant Axial Offset Control~~) (~~CAOC~~) Methodology)

PDC-3 Axial Offset Control

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

The operating PDC-3 scheme used to control the axial power distribution, ~~CAOC~~, involves maintaining the AFD within a tolerance band around a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during unit maneuvers.

The target flux difference is determined at equilibrium xenon conditions. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., ≥ 210 steps withdrawn) for steady state operation at high power levels. The power level should be as near RTP as practical. The value of the target flux difference obtained under these conditions divided by the Fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.

The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QPTR LCOs limit the radial component of the peaking factors.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations. The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

PDC-3 axial offset Control

The ~~CAOC~~ methodology (Ref~~s~~ 1~~, 2, and 3~~) entails:

a. Establishing an envelope of allowed power shapes and power densities;

b. Devising an operating strategy for the cycle that maximizes unit flexibility (maneuvering) and minimizes axial power shape changes;

c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics; and

d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also limit the range of power distributions that are assumed as initial conditions in analyzing Condition 2, 3, and 4 events. This ensures that fuel cladding integrity is maintained for these postulated accidents. ~~The most important Condition 4 event is the loss of coolant accident. The most significant Condition 3 event is the loss of flow accident. The most significant Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents. Condition 2 accidents, assumed to begin from within the AFD limits, are used to confirm the adequacy of Overpower ΔT and Overtemperature ΔT trip setpoints.~~ 13

In Chapter 15 of the UFSAR.

The limits on the AFD satisfy Criterion 2 of the NRC Policy Statement.

(continued)

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator, through either the manual operation of the control banks, or automatic motion of control banks responding to temperature deviations resulting from either manual operation of the Chemical and Volume Control System to change boron concentration, or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 4). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

Part A of this LCO is modified by a Note that states the conditions necessary for declaring the AFD outside of the target band. The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire cycle or more than one band, each to be followed for a specific range of cycle burnup.

The target bands are defined in the COLR.

With THERMAL POWER ≥ 90% RTP, the AFD must be kept within the target band. With the AFD outside the target band with THERMAL POWER ≥ 90% RTP, the assumptions of the accident analyses may be violated.

or 0.9 APL whichever is less

Parts B and C of this LCO are modified by Notes that describe how the cumulative penalty deviation time is calculated. It is intended that the unit is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is ≥ 50% RTP and < 90% RTP (i.e., Part B of this LCO), a 1 hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours, is allowed during which the unit may

or 0.9 APL whichever is less

(continued)

BASES

LCO (continued)

be operated outside of the target band but within the acceptable operation limits provided in the COLR. This penalty time is accumulated at the rate of 1 minute for each 1 minute of operating time within the power range of Part B of this LCO (i.e., THERMAL POWER > 50% RTP but < 90% RTP or APL whichever is less). The cumulative penalty time is the sum of penalty times from Parts B and C of this LCO.

INSERT B 3.2.3-1

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part C of this LCO), deviations of the AFD outside of the target band are less significant. The accumulation of 1/2 minute penalty deviation time per 1 minute of actual time outside the target band reflects this reduced significance. With THERMAL POWER < 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than 50% RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

The frequency of monitoring the AFD by the unit computer is once per minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time.

Violating the LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its limits.

Figure B 3.2.3A-1 shows a typical target band and typical AFD acceptable operation limits.

APPLICABILITY

AFD requirements are applicable in MODE 1 above 15% RTP. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 3).

Between 15% RTP and 100% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions.

(continued)

INSERT IB3.2.3-1

Part B of the LCO is modified by a Note that describes the relationship of Allowable Power Level (APL) to RTP as a function of the heat flux hot channel factor at RTP, $F_Q^{RTP}(Z)$. The reactor core AFD is analyzed to 100% RTP or 100% APL, whichever is less. When $F_Q^V(Z)$ is less than its limits, 100% RTP is more limiting than 100% APL. When $F_Q^V(Z)$ is greater than its limits, 100% APL is more limiting than 100% RTP. Hence the APL results in a more restrictive operating envelope for AFD when $F_Q^V(Z)$ is greater than its limits.

BASES

APPLICABILITY (continued)

At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. The value of the AFD in these conditions does not affect the consequences of the design basis events.

For surveillance of the power range channels performed according to SR 3.3.1.6, deviation outside the target band is permitted for 16 hours and no penalty deviation time is accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 days.

Low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP.

ACTIONS

A.1

With the AFD outside the target band and THERMAL POWER ≥ 90% RTP or 0.9 APL whichever is less, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time.

B.1

If the AFD cannot be restored within the target band, then reducing THERMAL POWER to < 90% RTP or 0.9 APL whichever is less places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.

The allowed Completion Time of 15 minutes provides an acceptable time to reduce power to < 90% RTP or 0.9 APL whichever is less without allowing the plant to remain in an unanalyzed condition for an extended period of time.

C.1 and C.2

With THERMAL POWER < 90% RTP but ≥ 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour

(continued)

BASES

ACTIONS

C.1 and C.2 (continued) 26

if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of xenon buildup and decay. The reduction to a power level < 50% RTP puts the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter. INSERT B3.23-2 26

If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. (Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits.) The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.

Condition C is modified by a Note that requires that Required Actions C.1 and C.2 must be completed whenever this Condition is entered.

D.1

If Required Action C.1 is not completed within its required Completion Time of 30 minutes, the axial xenon distribution starts to become significantly skewed with the THERMAL POWER ≥ 50% RTP. In this situation, the assumption that a cumulative penalty deviation time of 1 hour or less during the previous 24 hours while the AFD is outside its target band is acceptable at < 50% RTP, is no longer valid.

Reducing the power level to < 15% RTP within the Completion Time of 9 hours and complying with LCO penalty deviation time requirements for subsequent increases in THERMAL POWER ensure that acceptable xenon conditions are restored.

This Required Action must also be implemented either if the cumulative penalty deviation time is > 1 hour during the

(continued)

INSERT IB3.2.3-2

Restoration of cumulative penalty time to less than one (1) hour prior to increasing THERMAL POWER to above ≥ 50% RTP in accordance with Required Action C.2 ensures that the AFD will be within the core analysis.

2

BASES

ACTIONS

D.1 (continued)

previous 24 hours, or the AFD is not within the target band and not within the acceptable operation limits.

Condition D is modified by a Note that requires Action D.1 be completed whenever this Condition is entered.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

15

The AFD is monitored on an automatic basis using the unit process computer that has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFDs for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is > 90% RTP or 0.9 APL, whichever is less. During operation at THERMAL POWER levels < 90% RTP but > 15% RTP, the computer sends an alarm message when the cumulative penalty deviation time is > 1 hour in the previous 24 hours.

This Surveillance verifies that the AFD as indicated by the NIS excore channels is within the target band and consistent with the status of the AFD monitor alarm. The Surveillance Frequency of 7 days is adequate because the AFD is controlled by the operator and monitored by the process computer. Furthermore, any deviations of the AFD from the target band that is not alarmed should be readily noticed.

15

SR 3.2.3.2

With the AFD monitor alarm inoperable, the AFD is monitored to detect operation outside of the target band and to compute the penalty deviation time. During operation at ≥ 90% RTP or 0.9 APL, whichever is less, the AFD is monitored at a Surveillance Frequency of 15 minutes to ensure that the AFD is within its limits at high THERMAL POWER levels. At power levels < 90% RTP but > 15% RTP, the Surveillance Frequency is reduced to 1 hour because the AFD may deviate from the target band for up to 1 hour using the methodology of Parts B and C of this LCO to calculate the cumulative penalty deviation time before corrective action is required.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.2 (continued)

SR 3.2.3.2 is modified by a Note that states that monitored and logged values of the AFD are assumed to exist for the preceding 24 hour interval in order for the operator to compute the cumulative penalty deviation time. The AFD should be monitored and logged more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

~~SR 3.2.3.3~~

~~This Surveillance requires that the target flux difference is updated at a Frequency of 31 effective full power days (EFPD) to account for small changes that may occur in the target flux differences in that period due to burnup by performing SR 3.2.3.4.~~

~~Alternatively, linear interpolation between the most recent measurement of the target flux differences and a predicted end of cycle value provides a reasonable update because the AFD changes due to burnup tend toward 0% AFD. When the predicted end of cycle AFD from the cycle nuclear design is different from 0%, it may be a better value for the interpolation.~~ 18

SR 3.2.3.~~4~~ 3

Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux map provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

A Frequency of 31 EFPD after each refueling and ~~92~~ 31 EFPD thereafter for remeasuring the target flux differences adjusts the target flux difference for each excore channel to the value measured at steady state conditions. ~~This is the basis for the CAOC. Remeasurement at this Surveillance interval also establishes the AFD target flux difference~~ 18 18

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.4 (continued)

values that account for changes in incore excore calibrations that may have occurred in the interim.

A Note modifies this SR to allow the predicted end of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.

INSERT 1B3.2.3-3

REFERENCES

1. WCAP-8403 (nonproprietary). "Power Distribution Control and Load Following Procedures." Westinghouse Electric Corporation. September 1974.

2. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC). Attachment: "Operation and Safety Analysis Aspects of an Improved Load Follow Package." January 31, 1980.

3. C. Eicheldinger to D. B. Vassallo (Chief of Light Water Reactors Branch, NRC). Letter NS-CE-687. July 16, 1975.

4. FSAR, Chapter [15].

INSERT 1B3.2.3-4

INSERT IB3.2.3-3

A second Note modifies this SR to require that the target flux difference be determined in conjunction with the measurement of the heat flux hot channel factor, $F_Q(Z)$, in accordance with SR 3.2.1.1. This is a requirement of the PDC-3 Axial Offset Control Methodology.

INSERT IB3.2.3-4

REFERENCES

1. ANF-88-054 (Proprietary). "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," Advanced Nuclear Fuels Corporation, Richland, WA 99352, July 1988 (Submitted to NRC by CP&L letter dated August 24, 1989).

2. UFSAR Section 7.2.1.1

3. XN-NF-77-57(P)(A) (Proprietary). "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II," Supplement 2 and Supplement 2, Addendum 1," Exxon Nuclear Company, Richland WA 99352, October 1982, page 34.

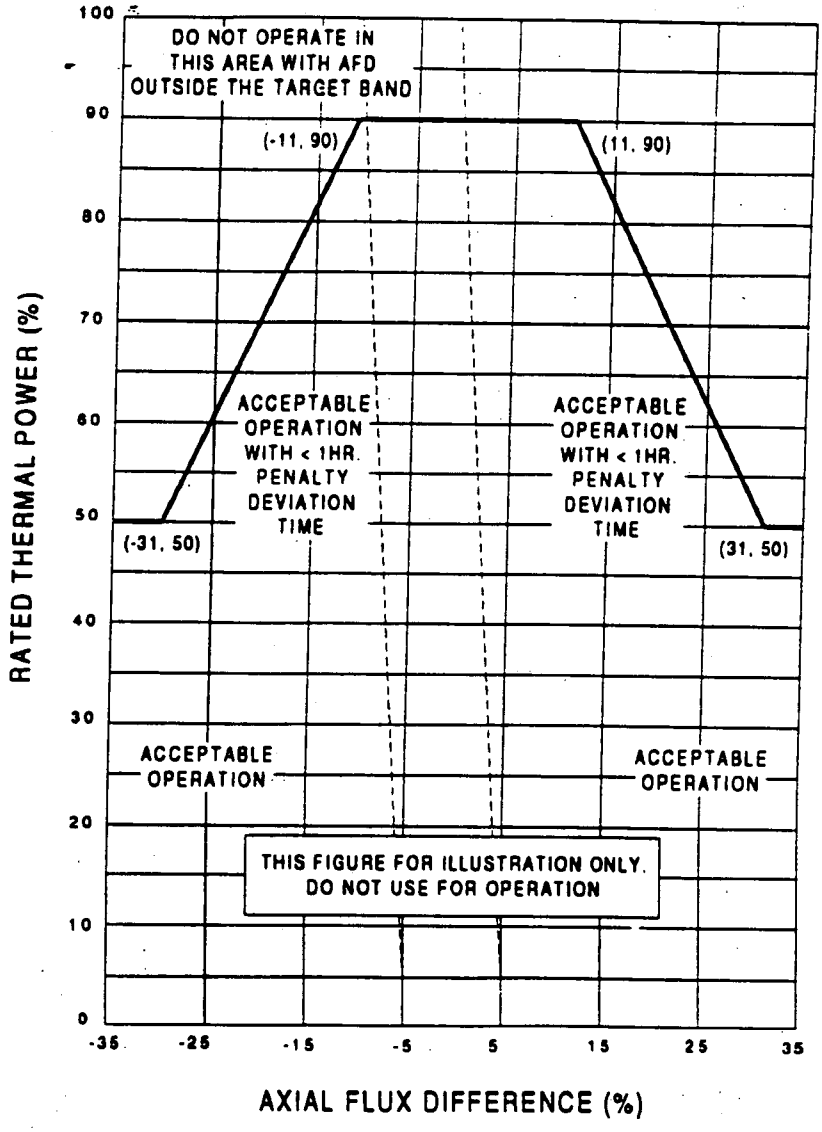


Figure B 3.2.3A-1 (Page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits and Target Band Limits as a Function of RATED THERMAL POWER

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3B AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

RAOC is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

Although the RAOC defines limits that must be met to satisfy safety analyses, typically an operating scheme, Constant Axial Offset Control (CAOC), is used to control axial power distribution in day to day operation (Ref. 1). CAOC requires that the AFD be controlled within a narrow tolerance band around a burnup dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.

The CAOC operating space is typically smaller and lies within the RAOC operating space. Control within the CAOC operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The RAOC methodology (Ref. 2) establishes a xenon distribution library with tentatively wide AFD limits. One dimensional axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA and loss of flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition 2, 3, or 4 events. This ensures that the fuel cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the LOCA. The most important Condition 3 event is the loss of flow accident. The most important Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents. Condition 2 accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of the NRC Policy Statement.

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks or automatic motion of control banks. The automatic motion

(continued)

BASES

LCO (continued)

of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 3). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

The AFD limits are provided in the COLR. Figure B 3.2.3B-1 shows typical RAOC AFD limits. The AFD limits for RAOC do not depend on the target flux difference. However, the target flux difference may be used to minimize changes in the axial power distribution.

Violating this LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its specified limits.

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.

ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion Time of

(continued)

BASES

ACTIONS

A.1 (continued)

30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits and is consistent with the status of the AFD monitor alarm. With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored. With the AFD monitor alarm OPERABLE, the Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.

2. R. W. Miller et al., "Relaxation of Constant Axial Offset Control: F_Q Surveillance Technical Specification," WCAP-10217(NP), June 1983.

3. FSAR, Chapter [15].

AFD (RAOC Methodology)
B 3.2.3B

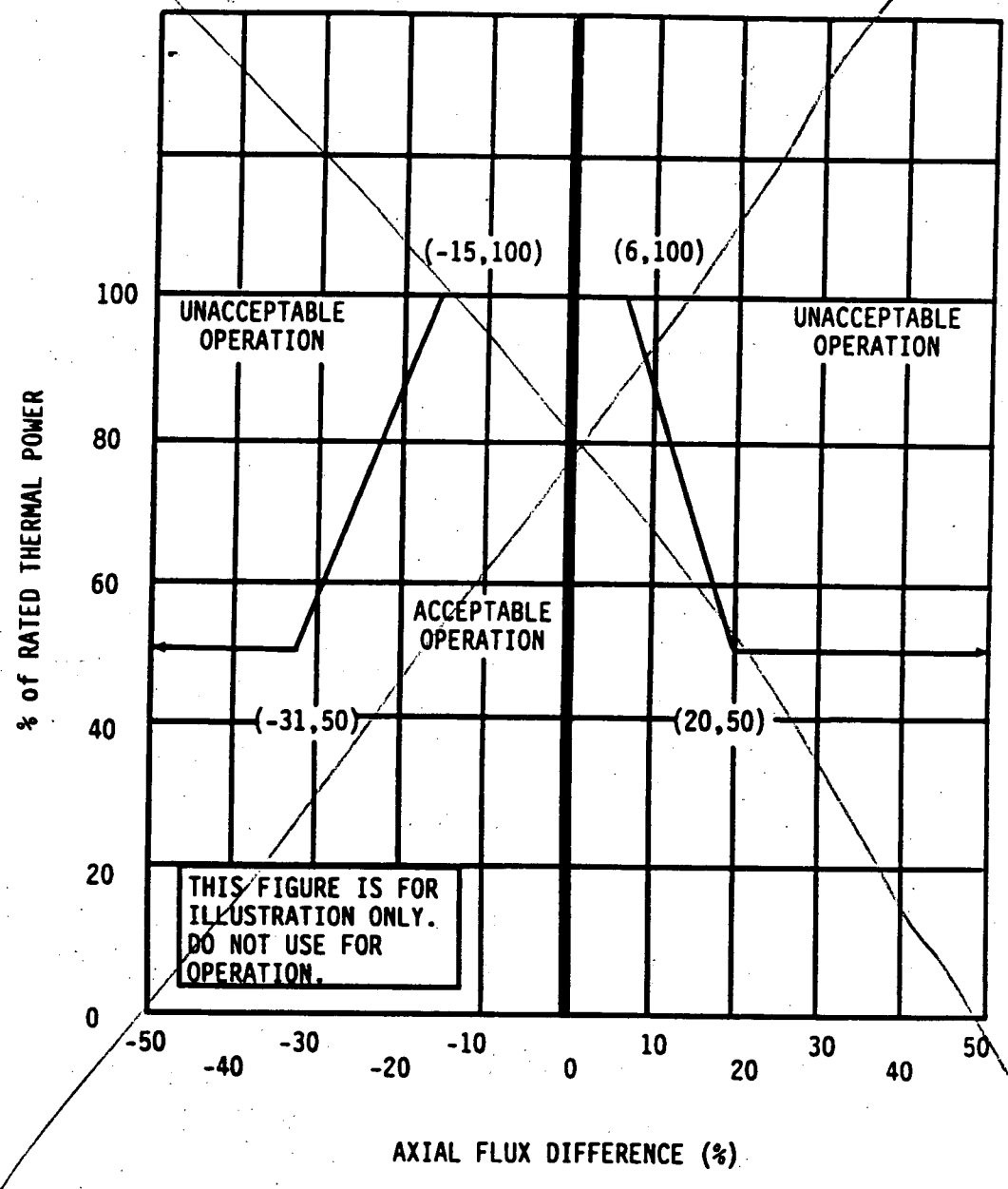


Figure B 3.2.3B-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
as a Function of RATED THERMAL POWER

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.8 "Control Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref.): and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref.).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and ($F^N_{\Delta H}$) is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient

(continued)

BASES

ACTIONS

A.1 (continued)

time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded

(continued)

2

BASES

ACTIONS

A.4 (continued)

and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

27

INSERT B 3.2.4-1

A.5

normalized to eliminate the indicated tilt

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are ~~recalibrated to show a zero QPTR~~ prior to increasing THERMAL POWER to above the limit of Required Action A.1 and A.2. This is done to detect any subsequent significant changes in QPTR.

19

20

Required Action A.5 is modified by a Note that states that the ~~QPTR is not zeroed out~~ indicated tilt is not eliminated until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). This Note is intended to prevent any ambiguity about the required sequence of actions.

A.6

excore detectors are normalized to eliminate the indicated tilt

19

Once the ~~QPTR is zeroed out~~ (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of reaching RTP. As an added precaution, if the

(continued)

INSERT IB3.2.4-1

Should Required Actions A.1, A.2, and A.3 result in restoration of QPTR within its limit, LCO 3.2.4 is satisfied, and Condition A can be exited prior to completion of Required Action A.4.

BASES

ACTIONS

A.6 (continued)

core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the ascent to power was begun. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1 and A.2, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been ~~calibrated to show zero~~ normalized to remove the tilt (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are ~~calibrated to show zero~~ normalized to remove the tilt and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1 if more than one input from Power Range Neutron Flux channels are inoperable.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, or Emergency Response Facility Information System (ERFIS) is

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 (continued)

within its limits. The Frequency of 7 days when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.

When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is required only when the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is ≥ 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. ~~The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8 for three and four loop cores.~~ 24

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued)

core flux map, to generate an incore QPTR. Therefore, the incore QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 4.4.2.1
3. ~~Regulatory Guide 1.77, Rev 01, May 1974~~ UFSAR Section 5.4.8
4. ~~10 CFR 50, Appendix A, GDC 26~~ UFSAR Section 3.1.3

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.2 - POWER DISTRIBUTION LIMITS

PART 7

JUSTIFICATION FOR DIFFERENCES (JFDs) TO ISTS BASES

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain bases sections of NUREG-1431, "Improved Standardized Technical Specifications - Westinghouse Plants," Revision 1, (ISTS) are not incorporated into ITS because they are not applicable. Specifically, the bases to ISTS Section 3.2.1A, "Heat Flux Hot Channel Factor (F_Q (Z)) (F_{xy} Methodology), " and Section 3.2.3B, "Axial Flux Difference (AFD) (Relaxed Axial Offset Control (RAOC) Methodology), are not applicable to the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No 2, and are not included in the ITS bases.

2 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted in the bases which do not result in technical changes (either actual or interpretational). Editorial changes which involve the insertion of plant specific terms or parameters are used to preserve consistency with the CTS and licensing basis.

3 Bases 3.2.1 and 3.2.3 are modified to reflect the PDC-3 axial offset control methodology utilized for fuel manufactured by Siemens Power Corporation.

4 Bases 3.2.1 are modified to replace the variable expression $F_Q^W(Z)$ with $F_Q^V(Z)$, to describe the relationship of $F_Q^V(Z)$ to $F_Q^C(Z)$ and $F_Q(Z)$, and to provide other clarifications throughout in order to reflect the PDC-3 axial offset control methodology.

5 Bases 3.2.1 are modified to add a paragraph to describe the addition of ITS 3.2.1 Required Action A.1, with subsequent renumbering, and to change the required Completion Time for Required Action A.2 to 30 minutes in order to maintain consistency with Required Action A.1.

6 Bases 3.2.1 are modified to add the Overtemperature ΔT (OTΔT) trip function to be consistent with the addition of the OTΔT function to Required Action A.3 in the ITS.

7 Bases 3.2.1 are modified to delete ISTS 3.2.1 Required Action B.1, with subsequent renumbering, and ISTS SR 3.2.1.1, with subsequent renumbering, to be consistent with the ITS and the PDC-3 axial offset control methodology.

8 Bases 3.2.1 are modified to reflect the PDC-3 axial offset control methodology, which applies only to the middle 80% of the core.

9 Bases 3.2.1 are modified to be consistent with PDC-3 axial offset control methodology, which utilizes $F_{\Delta H}$ to trend power distribution limits rather than $F_Q^C(Z)$.

10 HBRSEP was designed and licensed to the proposed Appendix A to 10 CFR 50, which was published in the Federal Register on July 11, 1967 (32FR10213). Appendix A to 10 CFR 50, which became effective in 1971, and was subsequently amended, is somewhat different from the proposed 1967 criteria. Updated Final Safety Analysis Report (UFSAR) Section 3.1 includes an evaluation of HBRSEP with respect to the proposed 1967 criteria. ISTS statements concerning the general design criteria are modified in the ITS to reference the current licensing basis description in UFSAR Section 3.1.

11 Bases 3.2.2 are modified to be consistent with the PDC-3 axial offset control methodology which allows only an increase of 0.2% in the $F_{\Delta H}$ limits for every 1% that THERMAL POWER is reduced below RATED THERMAL POWER (RTP).

12 Bases 3.2.2 are modified to add a Note to Surveillance Requirement (SR) 3.2.2.1. The Note describes the PDC-3 axial offset control methodology for trending $F_{\Delta H}$. If $F_{\Delta H}$ is increasing, a penalty is applied to the heat flux hot channel factor, or an increased Surveillance Frequency of the heat flux hot channel factor and target flux differences is required, until $F_{\Delta H}$ no longer indicates an increasing trend.

13 Bases 3.2.3 are modified to delete references to the sensitivity of specific accidents to axial flux difference, and Chapter 15 of the UFSAR is referenced for specific accident analysis.

14 Bases 3.2.3 are modified to incorporate the two distinct target bands for operation, i.e., ±3% and ±5%, from the PDC-3 axial offset control methodology. These bands and their corresponding V(Z) curves are defined in the CORE OPERATING LIMITS REPORT (COLR).

15 Bases 3.2.3 are modified to incorporate the Allowable Power Level (APL) restriction from the PDC-3 axial offset control methodology. The APL restriction requires that the AFD limitation curves be adjusted downward when the APL is less than 100%.

16 Bases 3.2.3 are modified to include Note 2 to LCO 3.2.3.b that describes Allowable Power Level (APL).

17 Bases 3.2.3 are modified to clarify that THERMAL POWER levels less than 50% RTP can affect the power distribution as power increases.

18 Bases 3.2.3 are modified by deleting SR 3.2.3.3, and subsequently renumbering the remaining SR. The Bases are also modified by changing the Frequency of the new SR 3.2.3.3 to 31 Effective Full Power Days (EFPDs) from 92 EFPDs. The PDC-3 axial offset control methodology does not allow the use of linear interpolation to determine the target flux values.

19 Bases 3.2.4 are modified to state that the excore detectors are normalized to eliminate the tilt rather than calibrated to show a zero tilt in ITS 3.2.4 Required Actions A.5 and A.6. This action is performed once the safety analyses requirements have been determined to be met assuming the indicated tilt.

20 Bases 3.2.4 are modified to include applicability of ITS 3.2.4 Required Action A.2 to the Completion Time for Required Actions A.5 and A.6, to reflect that THERMAL POWER limitations from either Required Action A.1 or A.2 may be more limiting.

21 Bases 3.2.4 are modified in Note 2 to SR 3.2.4.1 to allow the performance of SR 3.2.4.2 in lieu of SR 3.2.4.1 at any time. Since the verification of QUADRANT POWER TILT RATIO (QPTR) is most accurately performed with the incore detector system, performance of 3.2.4.2 is the preferred method to verify QPTR.

22 Bases 3.2.4 are modified in SR 3.2.4.1 to allow the verification of QPTR by calculation, or as indicated by the Emergency Response Facility Information System (ERFIS) computer. The ERFIS performs the QPTR calculation utilizing direct inputs from plant instrumentation.

23 Not used.

24 Bases 3.2.4 are modified in SR 3.2.4.2 to delete details of the incore system utilized for determining the QPTR in the ISTS reference plant and are not applicable to HBRSEP, Unit No. 2.

25 Bases 3.2.3 are modified to correct applicability for LCO 3.2.3 to 15% RTP through 100% RTP.

26 Bases 3.2.3 are modified to add Required Action C.2, which provides consistency with analyses performed in accordance with the PDC-3 axial offset control methodology.

27 Bases 3.2.4 are modified to provide clarification for Required Action A.4 in the event that LCO 3.2.4 is satisfied during performance of the Required Action.

28 Bases 3.2.3 is modified by adding Note 2 to ITS SR 3.2.3.3 bases which states that the target flux difference be determined in conjunction with

the measurement of the heat flux hot channel factor, $F_Q(Z)$, in accordance with ITS SR 3.2.1.1. The performance of SR 3.2.3.3. in conjunction with SR 3.2.1.1 is a requirement of the PDC-3 axial offset control methodology.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.2 - POWER DISTRIBUTION LIMITS

PART 8

PROPOSED HBRSEP, UNIT NO. 2 ITS

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

LCO 3.2.1 $F_Q(Z)$, as approximated by $F_Q^V(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. $F_Q^V(Z)$ not within limit.	A.1	Reduce AFD target band limits to restore $F_Q^V(Z)$ to within limit.	15 minutes
	OR		
	A.2.1	Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^V(Z)$ exceeds limit.	30 minutes
	AND		
	A.2.2	Reduce Power Range Neutron Flux–High trip setpoints ≥ 1% for each 1% $F_Q^V(Z)$ exceeds limit.	8 hours
	AND		
	A.2.3	Reduce Overpower and Overtemperature ΔT trip setpoints ≥ 1% for each 1% $F_Q^V(Z)$ exceeds limit.	72 hours
	AND		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Perform SR 3.2.1.1.	Prior to increasing THERMAL POWER above the limit of Required Action A.2.1
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

----------NOTE----------
During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify $F_Q^V(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^V(Z)$ was last verified AND 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ---------- $F^N_{\Delta H}$ not within limit.	A.1.1 Restore $F^N_{\Delta H}$ to within limit.	4 hours
	OR	
	A.1.2.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND	
	A.1.2.2 Reduce Power Range Neutron Flux–High trip setpoints to ≤ 55% RTP.	8 hours
	AND	
	A.2 Perform SR 3.2.2.1.	24 hours
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 --------NOTE--------- THERMAL POWER does not have to be reduced to comply with this Required Action. ----------	
	Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP AND Prior to THERMAL POWER exceeding 75% RTP AND 24 hours after THERMAL POWER reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	----------NOTE---------- If $F^N_{\Delta H}$ is within limits and measurements indicate that $F^N_{\Delta H}$ is increasing with exposure then: a. Increase $F^V_Q(Z)$ by a factor of 1.02 and reverify $F^V_Q(Z)$ is within limits; or b. Perform SR 3.2.1.1 and SR 3.2.3.3 once per 7 EFPD until two successive measurements indicate $F^N_{\Delta H}$ is not increasing. ----------	
	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (PDC-3 Axial Offset Control Methodology)

LCO 3.2.3 The AFD:

a. Shall be maintained within the target band about the target flux difference. The target band is specified in the COLR.

----------NOTE----------
The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.
---

b. May deviate outside the target band with THERMAL POWER < 90% RTP or 0.9 APL, whichever is less, but ≥ 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR.

----------NOTES----------
1. Penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

2. The Allowable Power Level (APL) is the limitation placed on THERMAL POWER for the purposes of applying the AFD target flux and operational limit curves. The APL is as follows:

$$APL = \text{minimum over } Z \text{ of } (100\%)(F_Q^{RTP}(Z))(K(Z))/F_Q^{V}(Z)$$
---

c. May deviate outside the target band with THERMAL POWER < 50% RTP.

----------NOTE----------
Penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP.

----------NOTE----------
A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.6, provided AFD is maintained within acceptable operation limits.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ 90% RTP or 0.9 APL, whichever is less. AND AFD not within the target band.	A.1 Restore AFD to within target band.	15 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 90% RTP or 0.9 APL, whichever is less.	15 minutes

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Required Action C.1 and C.2 must be completed whenever Condition C is entered. ---------- THERMAL POWER < 90% RTP or 0.9 APL, whichever is less, and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours. <u>OR</u> THERMAL POWER < 90% RTP or 0.9 APL, whichever is less, and ≥ 50% RTP with AFD not within the acceptable operation limits.	C.1 Reduce THERMAL POWER to < 50% RTP. <u>AND</u> C.2 Restore cumulative penalty deviation time to less than 1 hour.	30 minutes Prior to increasing THERMAL POWER to ≥ 50% RTP
D. ---------NOTE--------- Required Action D.1 must be completed whenever Condition D is entered. ---------- Required Action and associated Completion Time for Condition C not met.	D.1 Reduce THERMAL POWER to < 15% RTP.	9 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD is within limits for each OPERABLE excore channel.	7 days
SR 3.2.3.2	----------NOTE---------- Assume logged values of AFD exist during the preceding time interval. ---------- Verify AFD is within limits and log AFD for each OPERABLE excore channel.	-----NOTE------ Only required to be performed if AFD monitor alarm is inoperable ---------- Once within 15 minutes and every 15 minutes thereafter when THERMAL POWER ≥ 90% RTP or 0.9 APL, whichever is less AND Once within 1 hour and every 1 hour thereafter when THERMAL POWER < 90% RTP or 0.9 APL, whichever is less

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.3.3	----------NOTE---------- 1. The initial target flux difference after each refueling may be determined from design predictions. 2. The target flux difference shall be determined in conjunction with the measurement of $F_Q(Z)$ in accordance with SR 3.2.1.1. ----------	
	Determine, by measurement, the target flux difference of each OPERABLE excore channel.	Once within 31 EFPD after each refueling AND 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. QPTR not within limit.	A.1	Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours
	AND		
	A.2	Perform SR 3.2.4.1 and reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	Once per 12 hours
	AND		
	A.3	Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours AND Once per 7 days thereafter
	AND		
	A.4	Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.5	--------NOTE--------- Perform Required Action A.5 only after Required Action A.4 is completed. ----------	
		Normalize excore detectors to show zero QPTR.	Prior to increasing THERMAL POWER above the limit of Required Action A.1 or A.2
	AND		
	A.6	--------NOTE--------- Perform Required Action A.6 only after Required Action A.5 is completed. ----------	
		Perform SR 3.2.1.1 and SR 3.2.2.1.	Within 24 hours after reaching RTP OR Within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1 or A.2
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	----------NOTES---------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. ---------- Verify QPTR is within limit by calculation.	7 days AND Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable.
SR 3.2.4.2	----------NOTE---------- Only required to be performed if input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER ≥ 75% RTP. ---------- Verify QPTR is within limit using the movable incore detectors.	Once within 12 hours AND 12 hours thereafter

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.2 - POWER DISTRIBUTION LIMITS

PART 9

PROPOSED BASES TO HBRSEP, UNIT NO. 2 ITS

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD) (PDC-3 Axial Offset Control Methodology)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(Z)$ that are present during nonequilibrium situations, such as load following.

To account for these possible variations, the steady state value of $F_Q(Z)$ is adjusted by an elevation dependent factor that accounts for the calculated worst case transient conditions.

Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of

(continued)

BASES

BACKGROUND (continued)

the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition (Ref. 2);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 3); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 4).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{CFQ}{P} K(Z) \quad \text{for } P > 0.5$$

$$F_Q(Z) \leq \frac{CFQ}{0.5} K(Z) \quad \text{for } P \leq 0.5$$

where: CFQ is the $F_Q(Z)$ limit at RTP provided in the COLR,

K(Z) is the normalized $F_Q(Z)$ as a function of core height provided in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

For this facility, the actual values of CFQ and K(Z) are given in the COLR; however, CFQ is normally a number on the order of 2.5, and K(Z) is a function that looks like the one provided in Figure B 3.2.1B-1.

For the PDC-3 constant axial offset control operation, $F_Q(Z)$ is approximated by $F_Q^C(Z)$ and $F_Q^V(Z)$. Thus, both $F_Q^C(Z)$ and $F_Q^V(Z)$ must meet the preceding limits on $F_Q(Z)$. $F_Q^V(Z)$ is the most limiting expression of $F_Q(Z)$. By requiring $F_Q^V(Z)$ to be within limits, $F_Q^C(Z)$ can be assured to be within limits in steady state and transient situations.

An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value ($F_Q^M(Z)$) of $F_Q(Z)$. Then,

$$F_Q^C(Z) = F_Q^M(Z)\ 1.0815$$

where 1.0815 is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.

$F_Q^C(Z)$ is an excellent approximation for $F_Q(Z)$ when the reactor is at the steady state power at which the incore flux map was taken.

(continued)

BASES

LCO (continued)

The expression for $F_Q^V(Z)$ is:

$$F_Q^V(Z) = F_Q^C(Z)\ V(Z)$$

where V(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. V(Z) is included in the COLR.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

The PDC-3 axial offset control methodology provides two distinct target bands for operation which are the ±3% and the ±5% target bands. The target band that is selected determines the V(Z) penalty to be applied in the calculation of $F_Q^V(Z)$. When operation is restricted to the ±3% target band, the V(Z) penalty is minimized, and the $F_Q^V(Z)$ is reduced. Thus when operation is restricted to the more restrictive target band, the result may be that $F_Q^V(Z)$ is within limits, and no reduction in THERMAL POWER is

(continued)

BASES

ACTIONS

A.1 (continued)

required. In the event that the reduced target band does not result in an acceptable $F_Q^V(Z)$, the THERMAL POWER will be reduced in accordance with Required Action A.2.1. The Completion Time of 15 minutes provides an acceptable time to reevaluate $F_Q^V(Z)$ within the more restrictive target band to determine if $F_Q^V(Z)$ remains within limits.

A.2.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^V(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^V(Z)$ is $F_Q^M(Z)$ multiplied by engineering uncertainty factors and the maneuvering penalty factor V(Z) as stated in the COLR. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 30 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

A.2.2

A reduction of the Power Range Neutron Flux–High trip setpoints by ≥ 1% for each 1% by which $F_Q^V(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 8 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.2.3

Reduction in the Overpower and Overtemperature ΔT trip setpoints by ≥ 1% for each 1% by which $F_Q^V(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

(continued)

BASES

ACTIONS (continued)

A.2.4

Verification that $F_Q^V(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 prior to increasing THERMAL POWER above the limit imposed by Required Action A.2.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

B.1

If Required Actions of Condition A are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 is modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_Q^V(Z)$ is within specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which it was last verified to be within specified limits. Because $F_Q^V(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^V(Z)$ is made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_Q^V(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

without verification of $F_Q^V(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which F_Q was last measured.

SR 3.2.1.1

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called V(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by V(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^V(Z)$.

The limit with which $F_Q^V(Z)$ is compared varies inversely with power and directly with the function K(Z) provided in the COLR.

The V(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F_Q^V(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 10% inclusive; and

b. Upper core region, from 90 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q(Z)$ limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

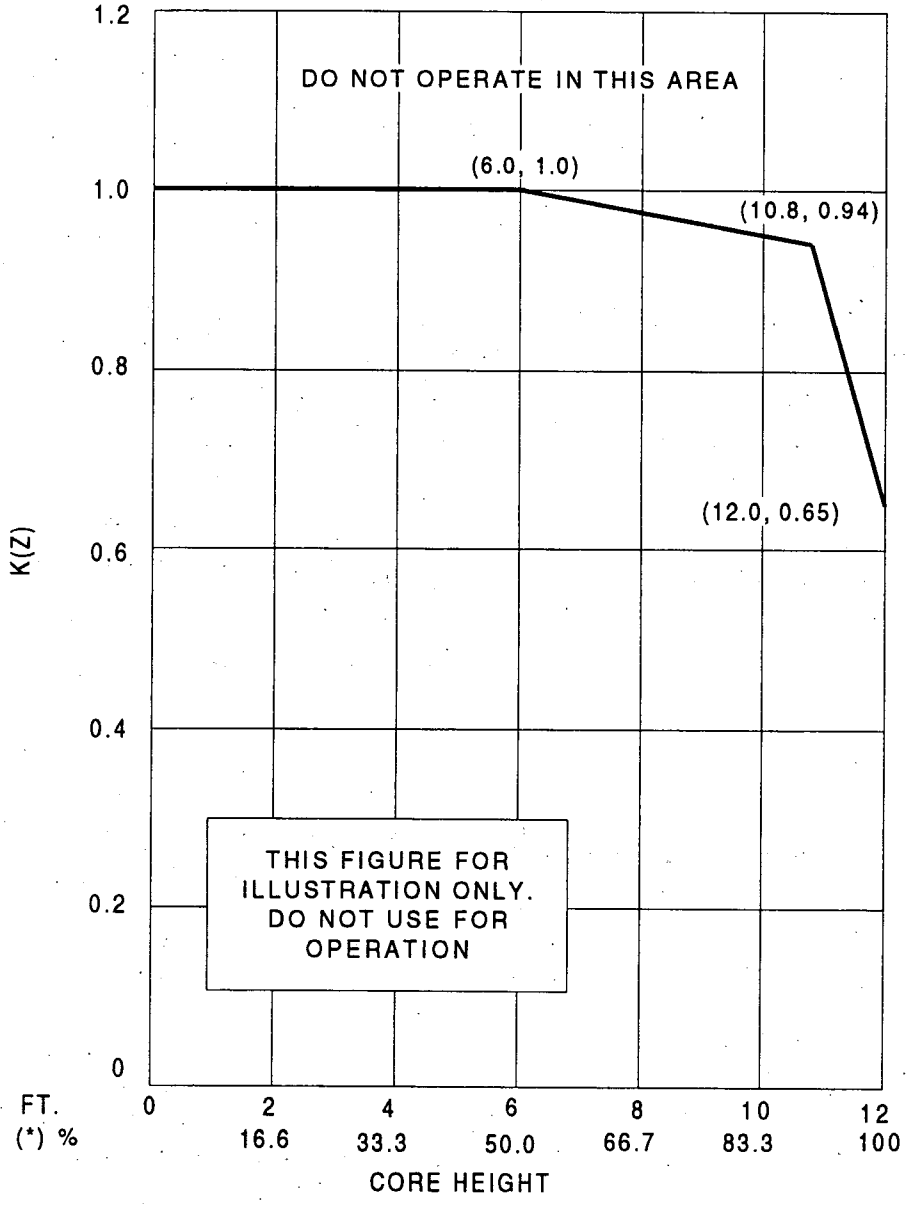
$F_Q(Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46, 1974.
2. UFSAR Section 4.4.2.1.
3. UFSAR Section 15.4.8.
4. UFSAR Section 3.1.



*For core height of 12 feet

Figure B 3.2.1B-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, F^N_{Δ} is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD) (PDC-3 Axial Offset Control Methodology)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to 1.154 using the Advanced Nuclear Fuels Corporation's DNB

(continued)

BASES

BACKGROUND (continued)

correlation (i.e., ANFP). All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements. Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition (Ref. 1);

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. Fuel design limits required by HBRSEP Design Criteria (Ref. 3) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion of 1.154 using the ANFP correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs)

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F^N_{\Delta H}$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 4).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD) (PDC-3 Axial Offset Control Methodology)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of the NRC Policy Statement.

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.

The limiting value of $F^N_{\Delta H}$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

(continued)

BASES

LCO (continued)

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F^N_{\Delta H}$ is allowed to increase 0.2% for every 1% RTP reduction in THERMAL POWER.

APPLICABILITY

The $F^N_{\Delta H}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F^N_{\Delta H}$ in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict $F^N_{\Delta H}$ in these modes.

ACTIONS

A.1.1

With $F^N_{\Delta H}$ exceeding its limit, the unit is allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time.

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, if power is not reduced because this Required Action is completed within the 4 hour time period, Required Action A.2 nevertheless requires another measurement and calculation of $F^N_{\Delta H}$ within 24 hours in accordance with SR 3.2.2.1.

(continued)

BASES

ACTIONS

A.1.1 (continued)

However, if power is reduced below 50% RTP, Required Action A.3 requires that another determination of $F^N_{\Delta H}$ must be done prior to exceeding 50% RTP, prior to exceeding 75% RTP, and within 24 hours after reaching or exceeding 95% RTP. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.

A.1.2.1 and A.1.2.2

If the value of $F^N_{\Delta H}$ is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux–High to ≤ 55% RTP in accordance with Required Action A.1.2.2. Reducing RTP to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 8 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System.

A.2

Once the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task

(continued)

BASES

ACTIONS

A.2 (continued)

over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

This Surveillance is modified by a Note that may require that the evaluation of $F^V_Q(Z)$ against its limits be performed with a penalty factor or that more frequent surveillances be performed. If $F^N_{\Delta H}$ is within limits and measurements indicate that $F^N_{\Delta H}$ is increasing with exposure, then $F^V_Q(Z)$ is increased by a factor of 1.02, and $F^V_Q(Z)$ is then reverified to be within limits: or, SR 3.2.1.1 and SR 3.2.3.3 are performed once per 7 EFPDs until two successive measurements of $F^N_{\Delta H}$ show that $F^N_{\Delta H}$ is not increasing. These alternative requirements prevent $F^V_Q(Z)$ from exceeding its limit for any significant period of time during the surveillance interval.

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

REFERENCES

1. UFSAR Section 4.4.2.1.
2. UFSAR Section 15.4.8.
3. UFSAR Section 3.1.
4. 10 CFR 50.46.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD) (PDC-3 Axial Offset Control Methodology)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

The operating scheme used to control the axial power distribution, PDC-3, involves maintaining the AFD within a tolerance band around a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during unit maneuvers.

The target flux difference is determined at equilibrium xenon conditions. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., ≥ 210 steps withdrawn) for steady state operation at high power levels. The power level should be as near RTP as practical. The value of the target flux difference obtained under these conditions divided by the Fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.

The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QPTR LCOs limit the radial component of the peaking factors.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations. The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The PDC-3 axial offset control methodology (Ref. 1) entails:

a. Establishing an envelope of allowed power shapes and power densities;

b. Devising an operating strategy for the cycle that maximizes unit flexibility (maneuvering) and minimizes axial power shape changes;

c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics; and

d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also limit the range of power distributions that are assumed as initial conditions in analyzing Condition 2, 3, and 4 events. This ensures that fuel cladding integrity is maintained for the postulated accidents in Chapter 15 of the UFSAR.

The limits on the AFD satisfy Criterion 2 of the NRC Policy Statement.

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator, through either the manual operation of the control banks, or automatic motion of control banks responding to

(continued)

BASES

LCO (continued)

temperature deviations resulting from either manual operation of the Chemical and Volume Control System to change boron concentration, or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 2). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

Part A of this LCO is modified by a Note that states the conditions necessary for declaring the AFD outside of the target band. The target bands are defined in the COLR.

With THERMAL POWER ≥ 90% RTP or 0.9 APL, whichever is less, the AFD must be kept within the target band. With the AFD outside the target band with THERMAL POWER ≥ 90% RTP or 0.9 APL, whichever is less, the assumptions of the accident analyses may be violated.

Parts B and C of this LCO are modified by Notes that describe how the cumulative penalty deviation time is calculated. It is intended that the unit is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is ≥ 50% RTP and < 90% RTP or 0.9 APL, whichever is less (i.e., Part B of this LCO), a 1 hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours, is allowed during which the unit may be operated outside of the target band but within the acceptable operation limits provided in the COLR. This penalty time is accumulated at the rate of 1 minute for each 1 minute of operating time within the power range of Part B of this LCO (i.e., THERMAL POWER > 50% RTP but < 90% RTP or 0.9 APL, whichever is less). The cumulative penalty time is the sum of penalty times from Parts B and C of this LCO.

(continued)

BASES

LCO (continued)

Part B of the LCO is modified by a Note that describes the relationship of Allowable Power Level (APL) to RTP as a function of the heat flux hot channel factor at RTP, $F_Q^{RTP}(Z)$. The reactor core AFD is analyzed to 100% RTP or 100% APL, whichever is less. When $F_Q^V(Z)$ is less than its limits, 100% RTP is more limiting than 100% APL. When $F_Q^V(Z)$ is greater than its limits, 100% APL is more limiting than 100% RTP. Hence the APL results in a more restrictive operating envelope for AFD when $F_Q^V(Z)$ is greater than its limits.

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part C of this LCO), deviations of the AFD outside of the target band are less significant. The accumulation of 1/2 minute penalty deviation time per 1 minute of actual time outside the target band reflects this reduced significance. With THERMAL POWER < 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than 50% RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

The frequency of monitoring the AFD by the unit computer is once per minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time.

Violating the LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its limits.

Figure B 3.2.3-1 shows a typical target band and typical AFD acceptable operation limits.

APPLICABILITY

AFD requirements are applicable in MODE 1 above 15% RTP. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 3).

(continued)

BASES

APPLICABILITY (continued)

Between 15% RTP and 100% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions.

At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. The value of the AFD in these conditions does not affect the consequences of the design basis events.

For surveillance of the power range channels performed according to SR 3.3.1.6, deviation outside the target band is permitted for 16 hours and no penalty deviation time is accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 days.

Low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP.

ACTIONS

A.1

With the AFD outside the target band and THERMAL POWER ≥ 90% RTP or 0.9 APL, whichever is less, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time.

B.1

If the AFD cannot be restored within the target band, then reducing THERMAL POWER to < 90% RTP or 0.9 APL, whichever is less places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.

The allowed Completion Time of 15 minutes provides an acceptable time to reduce power to < 90% RTP or 0.9 APL whichever is less without allowing the plant to remain in an unanalyzed condition for an extended period of time.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

With THERMAL POWER < 90% RTP or 0.9 APL, whichever is less but ≥ 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of xenon buildup and decay. The reduction to a power level < 50% RTP puts the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter. Restoration of cumulative penalty time to less than one (1) hour prior to increasing THERMAL POWER to above ≥ 50% RTP in accordance with Required Action C.2 ensures that the AFD will be within the core analysis.

If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. (Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits.) The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.

Condition C is modified by a Note that requires that Required Actions C.1 and C.2 must be completed whenever this Condition is entered.

D.1

If Required Action C.1 is not completed within its required Completion Time of 30 minutes, the axial xenon distribution starts to become significantly skewed with the THERMAL POWER ≥ 50% RTP. In this situation, the assumption that a cumulative penalty deviation time of 1 hour or less during the previous 24 hours while the AFD is outside its target band is acceptable at < 50% RTP, is no longer valid.

(continued)

BASES

ACTIONS

D.1 (continued)

Reducing the power level to < 15% RTP within the Completion Time of 9 hours and complying with LCO penalty deviation time requirements for subsequent increases in THERMAL POWER ensure that acceptable xenon conditions are restored.

This Required Action must also be implemented either if the cumulative penalty deviation time is > 1 hour during the previous 24 hours, or the AFD is not within the target band and not within the acceptable operation limits.

Condition D is modified by a Note that requires Action D.1 be completed whenever this Condition is entered.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The AFD is monitored on an automatic basis using the unit process computer that has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFDs for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is > 90% RTP or 0.9 APL, whichever is less. During operation at THERMAL POWER levels < 90% RTP or 0.9 APL, whichever is less but > 15% RTP, the computer sends an alarm message when the cumulative penalty deviation time is > 1 hour in the previous 24 hours.

This Surveillance verifies that the AFD as indicated by the NIS excore channels is within the target band and consistent with the status of the AFD monitor alarm. The Surveillance Frequency of 7 days is adequate because the AFD is controlled by the operator and monitored by the process computer. Furthermore, any deviations of the AFD from the target band that is not alarmed should be readily noticed.

SR 3.2.3.2

With the AFD monitor alarm inoperable, the AFD is monitored to detect operation outside of the target band and to compute the penalty deviation time. During operation at ≥ 90% RTP or 0.9 APL, whichever is less, the AFD is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.2 (continued)

monitored at a Surveillance Frequency of 15 minutes to ensure that the AFD is within its limits at high THERMAL POWER levels. At power levels < 90% RTP or 0.9 APL, whichever is less, but > 15% RTP, the Surveillance Frequency is reduced to 1 hour because the AFD may deviate from the target band for up to 1 hour using the methodology of Parts B and C of this LCO to calculate the cumulative penalty deviation time before corrective action is required.

SR 3.2.3.2 is modified by a Note that states that monitored and logged values of the AFD are assumed to exist for the preceding 24 hour interval in order for the operator to compute the cumulative penalty deviation time. The AFD should be monitored and logged more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

SR 3.2.3.3

Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux map provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

A Frequency of 31 EFPD after each refueling and 31 EFPD thereafter for remeasuring the target flux differences adjusts the target flux difference for each excore channel to the value measured at steady state conditions.

A Note modifies this SR to allow the predicted end of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.

A second Note modifies this SR to require that the target flux difference be determined in conjunction with the measurement of the heat flux hot channel factor, $F_Q(Z)$, in accordance with SR 3.2.1.1. This is a requirement of the PDC-3 Axial Offset Control Methodology.

(continued)

BASES (continued)

REFERENCES

1. ANF-88-054 (Proprietary). "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," Advanced Nuclear Fuels Corporation, Richland, WA 99352, July 1988 (Submitted to NRC by CP&L letter dated August 24, 1989).

2. UFSAR Section 7.2.1.1

3. XN-NF-77-57(P)(A) (Proprietary). "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II," Supplement 2 and Supplement 2, Addendum 1," Exxon Nuclear Company, Richland WA 99352, October 1982, page 34.

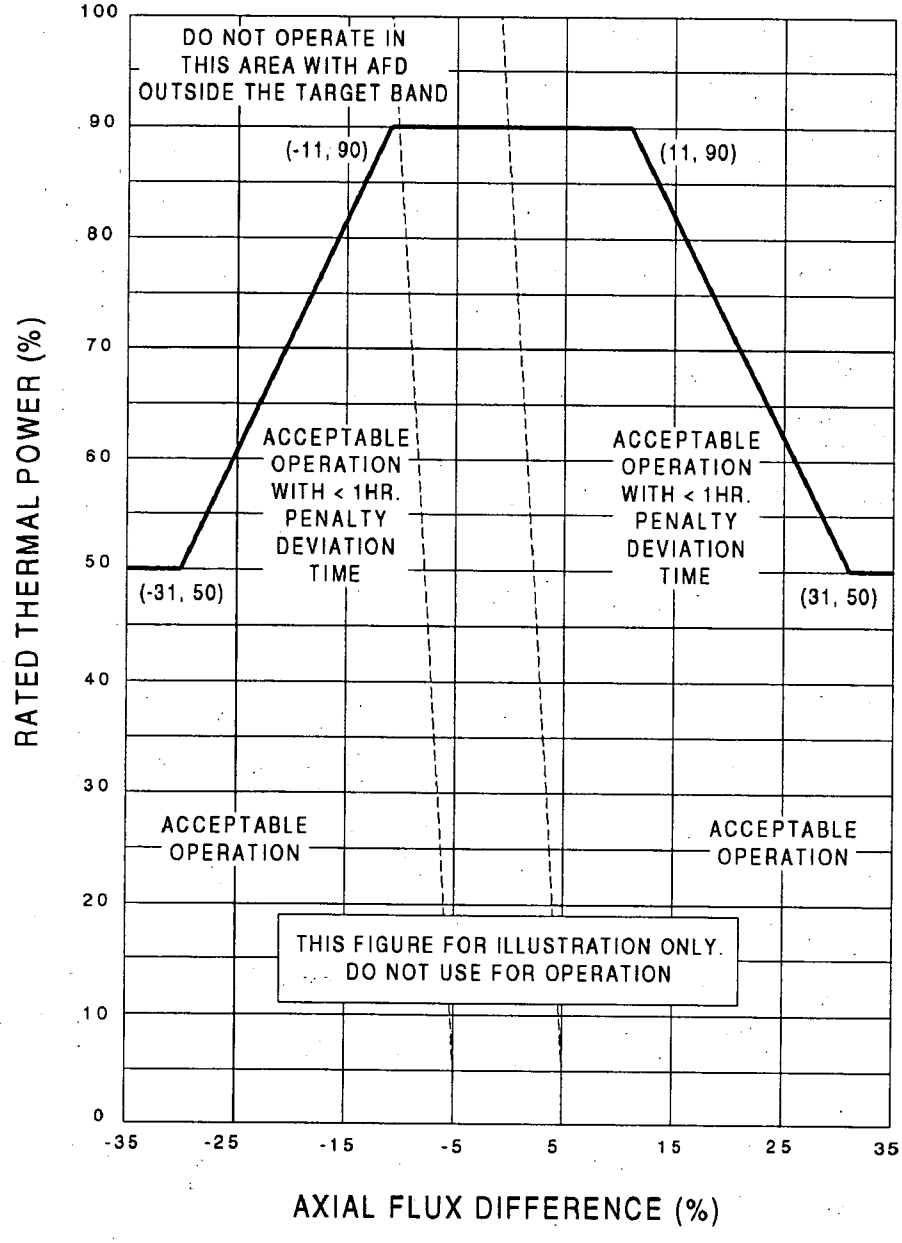


Figure B 3.2.3A-1 (Page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
and Target Band Limits as a Function
of RATED THERMAL POWER

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD) (PDC-3 Axial Offset Control Methodology)," LCO 3.2.4, and LCO 3.1.6, "Control Bank Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 2);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 3); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 4).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and ($F^N_{\Delta H}$) is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient

(continued)

BASES

ACTIONS

A.1 (continued)

time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded

(continued)

BASES

ACTIONS

A.4 (continued)

and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses. Should Required Actions A.1, A.2, and A.3 result in restoration of QPTR within its limit, LCO 3.2.4 is satisfied, and Condition A can be exited prior to completion of Required Action A.4.

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to eliminate the indicated tilt prior to increasing THERMAL POWER to above the limit of Required Action A.1 or A.2. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by a Note that states that the indicated tilt is not eliminated until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). This Note is intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the excore detectors are normalized to eliminate the indicated tilt (i.e., Required Action A.5 is performed), it

(continued)

BASES

ACTIONS

A.6 (continued)

is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of reaching RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the ascent to power was begun. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1 and A.2, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to remove the tilt (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are normalized to remove the tilt and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 (continued)

performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels or Emergency Response Facility Information System (ERFIS), is within its limits. The Frequency of 7 days when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.

When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is required only when the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is ≥ 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt.

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued)

core flux map, to generate an incore QPTR. Therefore, the incore QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

REFERENCES

1. 10 CFR 50.46.

2. UFSAR Section 4.4.2.1.

2. UFSAR Section 15.4.8.

3. UFSAR Section 3.1.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.2 - POWER DISTRIBUTION LIMITS

PART 10

ISTS GENERIC CHANGES

N/A

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
CONVERSION PACKAGE SECTION 3.3

ITS CONVERSION PACKAGE

CHAPTER 3.3 - INSTRUMENTATION

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.3 - INSTRUMENTATION

PART 1

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

ITS

A1

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, and flow and pressurizer level.

Objective

To provide for automatic protection action in the event that the principal process variables approach a safety limit.

Specification

[LCO 3.3.1] 2.3.1 Protective instrumentation settings for reactor trip shall be OPERABLE as follows:

2.3.1.1 Start-up protection

[T3.3.1-1 (2.b)] a. High flux, power range (low setpoint) ≤ 25% of rated power. 23.25

2.3.1.2 Core protection

[T3.3.1-1 (2.a)] a. High flux, power range (high setpoint) ≤ 109% of rated power 108

[T3.3.1-1 (7.b)] b. High pressurizer pressure ≤ 2385 psig. 2376

[T3.3.1-1 (7.a)] c. Low pressurizer pressure ≥ 1835 psig. 1844

M1

[T3.3.1-1 (5)] d. Overtemperature Δ T

[NOTE 1]

L1

The OTΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 2.96% of ΔT span.

$$\leq \Delta T_o \left\{ K_1 - K_2 \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} (T - T') + K_3 (P - P') - f(\Delta I) \right\}$$

M20

Add Trip Setpoints

Intermediate Range Neutron Flux	25% thermal power
Source Range Neutron Flux	1.0E5 cps
Steam Generator water level low	30%
Coincident with Steam Flow/Feedwater Flow Mismatch	6.4E5 lbm/hr
Turbine Trip low autostop oil pressure	45 psig

[T3.3.1-1(5)]
[NOTE 1]

where:

ΔT_o = Indicated ΔT at rated thermal power, °F;
T = Average temperature, °F;
P = Pressurizer pressure, psig;
K_1 = ~~≤ 1.1265~~; ≤ 1.1265 M1
K_2 = 0.01228;
K_3 = 0.00089;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation; LA1

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg}, τ_1 ≥ 20 seconds; τ_2 ≤ 3 seconds;

T′ ≤ 575.4°F Reference T_{avg} at rated thermal power; L2

P′ ≤ 2235 psig (Nominal RCS Operating Pressure);

S = Laplace transform operator, sec^{-1};

and f(ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant start-up tests such that:

1) For $(q_t - q_b)$ within +12% and -17%, where q_t and q_b are percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is total core power in percent of rated power (2300 Mwt), f(ΔI) = 0. For every 2.4% below rated power (2300 Mwt) level, permissible positive flux difference range is extended by +1 percent. For every 2.4% below rated power (2300 Mwt) level, the permissible negative flux difference range is extended by -1 percent. LA1

2) For each percent that the magnitude of $(q_t - q_b)$ exceeds +12% in a positive direction, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power (2300 Mwt)

2.4 $(q_T - q_b)$ − 12 percent A2

[T3.3.1-1 (5)] [NOTE 1]

(3) For each percent that the magnitude of (q_t - q_b) exceeds -17% in the negative direction, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power (2300 Mwt). A1

2.4 (q_b - q_T) - 17 percent A2

[T3.3.1-1 (6)] [NOTE 2]

e. Overpower Δ T

$$\leq \Delta T_o \left\{ K_4 - K_5 \left[\frac{\tau_3 S}{1 + \tau_3 S} \right] T - K_6 (T - T') - f(\Delta I) \right\}$$

The OPΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 3.17% of ΔT span. L1

where:

ΔT_o = Indicated ΔT at rated thermal power, °F;

T = Average temperature, °F;

T′ = 575.4°F Reference T_{avg} rated thermal power; M1

K_4 = 1.07; ≤ 1.06 L2

K_5 = 0.0 for decreasing average temperature, 0.02 sec/°F for increasing average temperature;

K_6 = 0.00277 for T > T′ and 0 for T ≤ T′;

S = Laplace transform operator, sec^{-1};

$\frac{\tau_3 S}{1 + \tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation; LA1

τ_3 = Time constant utilized in the rate-lag controller for T_{avg}, τ_3 ≤ 10 seconds; L2

f(ΔI) = As defined in d. above M1

[T3.3.1-1 (9)] f. Low reactor coolant loop flow ≥ 90% of normal indicated flow. 91

[T3.3.1-1 (12)] g. Low reactor coolant pump frequency ≥ 57.5 Hz. 58.2

[T3.3.1-1 (11)] h. Undervoltage ≥ 70% of normal voltage. 3120

a. Single loop
b. Two loops A3

2.3.1.3 Other Reactor Trips

[T3.3.1-1 (8)] a. High pressurizer water level ≤ 92% of span. 91 M1

[T3.3.1-1 (13)] b. Low-low steam generator water level ≥ 14% of narrow range instrument span. 16

2.3.2 Protective instrumentation settings for reactor trip interlocks shall be as follows:

[T 3.3.1-1] [Footnotes (f)(h)] 2.3.2.1 The low pressurizer pressure trip, high pressurizer level trip, and the low reactor coolant flow trip (for two or more loops) may be bypassed below 10% of rated power. P-7 P-8

[T 3.3.1-1] [Footnote (g)] 2.3.2.2 The single-loop-loss-of-flow trip may be bypassed below 45% of rated power.

2.3.3 The RCS narrow range temperature sensors response time shall be less than or equal to a 4.0 second lag time constant. LA2

Basis

The power range reactor trip low setpoint provides protection in the power range for a power excursion beginning from lower power. This trip value was used in the safety analysis.[1]

In the power range of operation, the overpower nuclear flux reactor trip protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed set point, with allowance for errors, is consistent with the trip point assumed in the accident analysis.[2]

The source and intermediate range reactor trips do not appear in the specification, as these settings are not used in the transient and accident analysis (FSAR Section 15). Both trips provide protection during reactor startup. The former is set at about 10^{+5} counts/sec and the latter at a current proportional to approximately 25% of full power.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is also a backup to the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss-of-coolant accident.[3]

A4

Specification 3.3.1

A1

ITS

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to plant operational safety instrumentation systems.

Objective

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification

3.5.1.1 The Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3.5-1. See 3.3.2

[ACTION A] [Applicability] 3.5.1.2 For on-line testing or in the event of a subsystem instrumentation channel failure, plant operation at rated power shall be permitted to continue in accordance with Tables ~~3.5-2 through 3.5-5~~ 3.3.1-1.

3.5.1.3 In the event the number of channels in service listed in Table 3.5-5 falls below the limits given in the column entitled Minimum Channels Operable, operation shall be limited according to the requirement shown in Column 2. See 3.3.3

3.5.1.4 The containment ventilation isolation function is only required when containment integrity is required. See 3.3.6

[ACTION A] 3.5.1.5 In the event the number of operable channels of a particular functional unit listed in Tables ~~3.5-2, 3, or 4~~ 3.3.1-1 falls below the limits given in the column entitled Total Number of Channels, operation shall be limited according to the requirement shown in Column 3. immediately

M2

Add ACTIONS "Note"

A5

~~Amendment No. 85~~

License Amendment Request, 12/10/95

A1

ITS

TABLE 3.5-2 (Continued)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

A27

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 MINIMUM CHANNELS OPERABLE	3 OPERATOR ACTION IF COLUMN 1 OR 2 CANNOT BE MET	APPLICABLE CONDITIONS
[T3.3.1-1(15)] 11.	Turbine Trip				
	A. Low Auto Stop Oil Pressure	3	2	ACTION 6	*****
	B. Turb Stop Valves	2	2	ACTION 6	*****
[T3.3.1-1(13)] 12.	Lo Lo Steam Generator Water Level	3/SG	2/SG	ACTION 6	MODES 1,2 Reactor Critical
[T3.3.1-1(12)] 13.	Underfrequency 4 kV System RCPs	3 1/bus	2	ACTION 6	MODE 1 (f) Reactor Critical
[T3.3.1-1(11)] 14.	Undervoltage OR 4 kV System	3 1/bus	2	ACTION 7	Reactor Critical
15.	Control Rod Misalignment Monitor				
	A. ERFIS Rod Position Deviation	1	1	ACTION 9	Reactor Critical
	B. Quadrant Power Tilt Monitor (upper and lower ex-core neutron detectors) "Detector Current Comparator"	1	1	ACTION 10	>50% of rated power

P E M M

L11 L10 L12 L4 LA7

Amendment No. 85

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Specification 3.3.1

A1

ITS

TABLE 3.5-2 (Continued)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 MINIMUM CHANNELS OPERABLE	3 OPERATOR ACTION IF COLUMN 1 OR 2 CANNOT BE MET	APPLICABLE CONDITIONS
[T3.3.1-1 (14)] 16.	Low Steam Generator Level Coincident With Steam Flow/Feedwater Flow Mismatch	2 Level and 2 Stm/ Feed Flow Mismatch Per SG	1 Level and 2 Stm/ Feed Flow Mismatch Per SG OR 2 Level and 1 Stm/ Feed Flow Mismatch Per SG	ACTION 6 E	MODES 1, 2 Reactor Critical

Amendment No.

License Amendment Request, 12/10/95

A1

ITS

TABLE 3.5-2 (Continued)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

TABLE NOTATIONS

(a) With the reactor trip breakers closed and rods not fully inserted or Rod Control system capable of rod withdrawal. L35

(b) Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) setpoint.

(d) Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoint.

(h) Above the ~~P-10 (Low Setpoint Power Range Neutron Flux Interlock) setpoint or~~ P-7 (Turbine First Stage Pressure Interlock) setpoint and below the P-8 (Low Setpoint Power Range Neutron Flux Interlock) setpoint.

(f) Above the ~~P-10 (Low Setpoint Power Range Neutron Flux Interlock) setpoint or~~ P-7 (Turbine First Stage Pressure Interlock) setpoint. L5

Add Notes (c), (e), (i) A26

ACTION STATEMENTS

[ACTION B] ~~ACTION 1~~ - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 ~~12~~ hours, or be in MODE 3 the Hot Shutdown Condition within 54 ~~the next 8~~ hours and open RTBs in 55 hours. L6

[ACTION D] ~~ACTION 2~~ - With the number of OPERABLE channels one less than the Total Number of Channels, Startup and/or Power Operation may proceed provided the following Conditions are satisfied: Add RA D.2.2 "NOTE" L7

a. The inoperable channel is placed in the tripped condition within 6 ~~1~~ hour.

b. Either, thermal power is restricted to less than or equal to 75% of rated power and the Power Range Neutron Flux trip setpoint is reduced to less than or equal to 85% of rated power within 4 hours; or, the Quadrant Power Tilt Ratio is monitored within 12 hours and every 12 hours thereafter, using the movable incore detectors to confirm that the normalized symmetric power distribution is consistent with the indicated Quadrant Power Tilt Ratio. L8

or, be in MODE 3 in 12 hours M6

~~ACTION 3~~ - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the thermal power level:

[ACTION H] a. Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoints, restore the inoperable channel to OPERABLE status prior to increasing thermal power above the P-6 setpoint.

[ACTION F] b. Above the P-6 (Intermediate Range Neutron Flux Interlock) setpoint but below 10% of rated power, ~~restore the inoperable channel to OPERABLE status prior to increasing thermal power above 10% of rated power.~~ M7

Reduce power to < P6 in 2 hrs or increase power to > P10 in 2 hrs

~~Amendment No.~~

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A1

ITS

TABLE 3.5-2 (Continued)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

TABLE NOTATIONS

[ACTION I] ~~ACTION 4~~ - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes. immediately M2

[ACTION L] ~~ACTION 5~~ - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with Shutdown Margin within 1 hour and at least once per 12 hours thereafter. Add RA L.1 and L.2 M3

[ACTION E] ~~ACTION 6~~ - With the number of OPERABLE channels one less than the Total Number of Channels, Startup and/or Power Operation may proceed ~~until performance of the next required operational test~~ provided the inoperable channel is placed into the tripped condition within ~~1~~ 6 hour or be in MODE 3 in 12 hours. L3 M4 L3

[ACTION M] ~~ACTION 7~~ - With the number of OPERABLE channels one less than the Total Number of Channels; place the inoperable channel into the tripped condition within ~~1~~ 6 hour ~~and restore the inoperable channel to OPERABLE status within 7 days or be in at least the Hot Shutdown Condition within the next 8 hours.~~ or reduce THERMAL POWER < P-7 in 12 hrs. L4

[ACTION N] → < P8 in 10 hrs

[ACTION P] < P7, 10 hrs

[ACTION C] ~~ACTION 8~~ - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip Breakers. in 49 hours. M5

[ACTION K]

~~ACTION 9 - Log individual rod position within 1 hour and every hour thereafter, and following load changes of >10% of rated power, or after >30 inches of control rod motion. In addition to the above ACTIONS, if both rod misalignment monitors (15.A and 15.B) are inoperable with reactor power >50% of rated power for 2 hours or more, the nuclear overpower trip shall be reset to ≤ 93% of rated power.~~

~~ACTION 10 Log individual upper and lower ion chamber currents within 1 hour and every hour thereafter, and following load changes of >10% of rated power, or after >30 inches of control rod motion. In addition to the above ACTIONS, if both rod misalignment monitors (15.A and 15.B) are inoperable with reactor power >50% of rated power for two hours or more, the nuclear overpower trip shall be reset to ≤ 93 percent of rated power.~~

LA7

~~Amendment No.~~

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A1

3.10.4 **Rod Drop Time**

3.10.4.1 The drop time of each control rod shall be not greater than 1.8 seconds at full flow and operating temperature from the beginning of rod motion to dashpot entry.

See 3.1.4

3.10.5 **Reactor Trip Breakers**

[T3.3.1-1 (18,19,20)] [Applicability] 3.10.5.1 The reactor shall not be made critical (MODES 1, 2) unless the following conditions are met:

A18

[T 3.3.1-1 (18,19)] a. Two reactor trip breakers (with undervoltage and shunt trip mechanisms) are operable.

LA3

b. ~~Reactor trip bypass breakers are racked out or removed.~~

[T3.3.1-1 (20)] c. Two trains of automatic trip logic are operable.

[ACTION Q] [ACTION R] 3.10.5.2 During power operation, the requirements of 3.10.5.1 may be modified to allow the following components to be inoperable. If the system is not restored to meet the requirements of 3.10.5.1, the reactor shall be placed in the hot shutdown (MODE 3) condition utilizing normal operating procedures within the next ~~8~~ 6 hours.

M8

[ACTION R] a. One reactor trip breaker may be inoperable for up to ~~12~~ 1 hours.

[ACTION Q] b. One train of automatic trip logic may be inoperable for up to ~~12~~ 6 hours.

[ACTION Q Note] [ACTION R Note 1] c. One reactor trip bypass breaker may be racked in and closed for up to 12 hours. for surveillance testing, provided the other train is OPERABLE.

M45

[ACTION U] 3.10.5.3 With one of the diverse trip features inoperable (shunt trip attachment/undervoltage trip attachment) on one of the reactor trip breakers, power operation may continue for up to 48 hours. If the

Add ACTION R Note 2

Add Table 3.3.1-1 Items 18, 19, 20 for MODE Applicability 3(a), 4(a), 5(a) and Required Actions V and C

M9

ITS

[T 3.3.1-1 (19)]
[ACTION U]

diverse trip feature is not restored to operable status within this time, the reactor shall be placed in ~~the hot shutdown condition~~ MODE 3 utilizing normal operating procedures within ~~the next eight~~ 54 hrs, and open RTB in 55 hrs hours. (A1) (M10)

3.10.6 Inoperable Control Rods

3.10.6.1 A control rod shall be deemed inoperable if (a) the rod is misaligned by more than 15 inches with its bank, (b) if the rod cannot be moved by its drive mechanism, or (c) if its rod drop time is not met.

See 3.1.4

Add ACTIONS J, S, T, V (M11)

Add ACTION G (L9)

Add Table 3.3.1-1 "ALLOWABLE VALUES" COLUMN (M12)

Add ACTION O (L37)

Add Table 3.3.3-1 Functions:

10. RCP Breaker Position
    a. Single Loop
    b. Two Loops
16. SI Input from ESFAS
17. RTS Interlocks
    a. Intermediate Range Neutron Flux
    b. P-7
    c. P-8
    d. P-10
    e. Turbine Impulse Pressure

(M13)

A1

TABLE 3.5-2

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

ITS

A27

	NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 MINIMUM CHANNELS OPERABLE	3 OPERATOR ACTION IF COLUMN 1 OR 2 CANNOT BE MET	APPLICABLE CONDITIONS
[T3.3.1-1 (1)]	1.	Manual	2	2	ACTION 1 (B)	MODES 1,2 Reactor Critical
			2	2	ACTION 8 (C)	Hot/Cold Shutdown (*) (a)
						MODES 3,4,5
[T3.3.1-1(2)]	2.	Nuclear Flux Power Range				
		A. High Setpoint	4	3	ACTION 2 (D)	MODES 1,2 Reactor Critical
		B. Low Setpoint	4	3	ACTION 2 (E)	Reactor Critical (**) (b)
[T3.3.1-1 (3)]	3.	Nuclear Flux Intermediate Range	2	2	ACTION 8 (F,G,H)	MODES 1,2 Reactor Critical (***) (c,d)
[T3.3.1-1 (4)]	4.	Nuclear Flux Source Range				
		A. Startup	2	2	ACTION 4 (I)	MODE 2 Reactor Critical (****) (d)
		B. Shutdown	2	1	ACTION 5 (L)	Hot/Cold Shutdown (e)
		C. Shutdown	2	2	ACTION 8 (K)	Hot/Cold Shutdown (*) (a)
						MODES 3,4,5
[T3.3.1-1(5)]	5.	Overtemperature ΔT	3	2	ACTION 6 (E)	Reactor Critical MODES 1,2
[T3.3.1-1(6)]	6.	Overpower ΔT	3	2	ACTION 6 (E)	Reactor Critical
[T3.3.1-1(7.a)]	7.	Low Pressurizer Pressure	3	2	ACTION 6 (M)	MODE 1 (f) *****
[T3.3.1-1(7.b)]	8.	Hi Pressurizer Pressure	3	2	ACTION 6 (E)	MODES 1,2 Reactor Critical
[T3.3.1-1(8)]	9.	Pressurizer Hi Water Level	3	2	ACTION 6 (M)	MODE 1 (f) *****
[T3.3.1-1(9)]	10.	Low Reactor Coolant Flow				
		A. Single Loop	3/loop	2/loop	ACTION 6 (N)	MODE 1 (g) >45% of rated power
		B. Two Loop	3/loop	2/loop	ACTION 6 (M)	**** MODE 1 (h)

A27

A6

L10

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Add SR "NOTE" A5

ITS

L13

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

L14

	Channel Description	Check	Calibrate	Test	Remarks
[T3.3.1-1(2,a)] 1.	Nuclear Power Range	S SR 3.3.1.1	Q (1) M (3) R (3) SR 3.3.1.2 SR 3.3.1.11	B/W (2) SR 3.3.1.7 SR 3.3.1.8	(1) Thermal Power calculations during power operations (2) Signal to ΔT; bistable action (permissive, rod stop, trips) (3) Upper and lower chambers for symmetric offset: monthly during power operations. When periods of reactor shutdown extend this interval beyond one month, the calibration shall be performed immediately following return to power.
[T3.3.1-1(3)] 2.	Nuclear Intermediate Range	S/N SR 3.3.1.1	N.A. SR 3.3.1.11	S/U (2) SR 3.3.1.8	(1) Once/shift when in service (2) Log level; bistable action (permissive, rod stop, trip)
[T3.3.1-1(4)] 3.	Nuclear Source Range	S/N SR 3.3.1.1	N.A. SR 3.3.1.11	S/U (2) SR 3.3.1.7 SR 3.3.1.8	(1) Once/shift when in service (2) Bistable action (alarm, trip)
T3.3.1-1(5,6)] 4.	Reactor Coolant Temperature OTΔT and OPΔT	S SR 3.3.1.1	R (4) SR 3.3.1.12 SR 3.3.1.3 SR 3.3.1.6	B/W (1)(2) SR 3.3.1.7 R (3)	(1) Overtemperature - ΔT (2) Overpower - ΔT (3) Narrow range RTD response time (4) To include narrow range RTD cross calibration
[T3.3.1-1(9)] 5.	Reactor Coolant Flow	S SR 3.3.1.1	R SR 3.3.1.10	M SR 3.3.1.7	
[T3.3.1-1(8)] 6.	Pressurizer Water Level	S SR 3.3.1.1	R SR 3.3.1.10	M SR 3.3.1.7	
[T3.3.1-1(7)] 7.	Pressurizer Pressure	S SR 3.3.1.1	R SR 3.3.1.10	M SR 3.3.1.7	
[T3.3.1-1(11)] 8.	Voltage RCP	N.A	R SR 3.3.1.10	M SR 3.3.1.9	Reactor Protection circuits only

L15 M15 A7 M16 A7 L14 M14 L16 A7

By means of the moveable in-core detector system

[T3.3.1-1(2.b)] Add SR 3.3.1.1, SR 3.3.1.8 and SR 3.3.1.11 for Power Range Neutron Flux-Low

Amendment No. 83, 88, 121

Add SRs 3.3.1.1, 3.3.1.2, 3.3.1.3, 3.3.1.4, 3.3.1.5, 3.3.1.6, 3.3.1.7, 3.3.1.8, 3.3.1.9, 3.3.1.10, 3.3.1.11, 3.3.1.12, 3.3.1.13, 3.3.1.14, and 3.3.1.15 A25

A1

Specification 3.3.1

ITS

TABLE 4.1-1 (Continued)
MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
9.	Analog Rod Position	S (1,2)	R	M	(1) With step counters (2) Following rod motion in excess of six inches when the computer is out of service
10.	Rod Position Bank Counters	S (1,2)	N.A.	N.A.	(1) Following rod motion in excess of six inches when the computer is out of service (2) With analog rod position
[T3.3.1-1 (13)] 11.	Steam Generator Level	S SR 3.3.1.1	R SR 3.3.1.10	M SR 3.3.1.7	
12.	Charging Flow	N.A.	R	N.A.	
13.	Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14.	Boric Acid Tank Level	D (1)	R	N.A.	(1) Bubbler tube rodded weekly
15.	Refueling Water Storage Tank Level	W	R	N.A.	
16.	Deleted				
17.	Volume Control Tank Level	N.A.	R	N.A.	
18.	Containment Pressure	D	R	B/W (1)	(1) Containment isolation valve signal
19.	Deleted by Amendment No. 85				
20.	Boric Acid Makeup Flow Channel	N.A.	R	N.A.	

LA4

L16

LA4

[T 3.3.1-1 (10)] Add SR 3.3.1.14 for RCP Breaker Position
[T3.3.1-1 (16)] Add SR 3.3.1.14 for SI input from ESFAS

M14

Specification 3.3.1

A1

ITS

TABLE 4.1-1 (Continued)

	Channel Description	Check	Calibrate	Test	Remarks
	21. Containment Sump Level	N.A.	R	N.A.	See 3.4.15
[T3.3.1-1(15)]	22. Turbine Trip Logic**	N.A.	~~N.A.~~ SR3.3.1.10	~~R~~ SR 3.3.1.15	M17
	23. Accumulator Level and Pressure	S	R	N.A.	
	24. Steam Generator Pressure	S	R	M	LA4
[T3.3.1-1(17.e)]	25. Turbine ~~First Stage~~ Impulse Pressure	~~S~~ SR3.3.1.1	~~R~~ SR3.3.1.10	~~M~~ SR3.3.1.13	L17
	26. ~~DELETED~~				
[T3.3.1-1(20)]	27. Logic Channel Testing Automatic Trip	N.A.	N.A.	M(1) S/U(2) SR 3.3.1.5	(1) Applicability MODES 1,2,3,4,5 During hot shutdown and power operations. ~~When periods of reactor cold shutdown and refueling extend this interval beyond one month, this test shall be performed prior to startup.~~ M18 M19
					(2) ~~Logic channel testing for nuclear source range channels shall only be required prior to each reactor startup, if not performed within the previous seven (7) days.~~ L18
	28. ~~DELETED~~				
[T3.3.1-1(12)]	29. ~~RV~~ RCPs Frequency	N.A.	~~R~~ SR3.3.1.10	~~R~~ SR3.3.1.14	

On a STAGGERED TEST BASIS

[T3.3.1-1(15)]** Stop valve closure or low EH fluid pressure.

[T 3.3.1-1 (17.a – d)] Add SR 3.3.1.11 and SR 3.3.1.13 For RPS interlocks P-6, P-7, P-8, P-10 M14

Specification 3.3.1 A1

ITS

TABLE 4.1-1 (Continued)

	Channel Description	Check	Calibrate	Test	Remarks
[T3.3.1-1(18,19)] 30.	Reactor Trip Breakers	N.A.	N.A..	M(1) SR 3.3.1.4	(1) ~~The reactor trip breaker trip actuating device operational test shall verify the operability of the UV trip attachment and the shunt trip attachment, individually.~~
31.	Overpressure Protection System	N.A.	R	M	See 3.4.12

L19

A7

Add SR 3.3.1.2 NOTES
SR 3.3.1.3 NOTES
SR 3.3.1.4 NOTE
SR 3.3.1.6 NOTE
SR 3.3.1.7 NOTE
SR 3.3.1.8 NOTES
SR 3.3.1.9 NOTE
SR 3.3.1.10 NOTE
SR 3.3.1.11 NOTE
SR 3.3.1.12 NOTE
SR 3.3.1.14 NOTE
SR 3.3.1.15 NOTE

A8

A1

Specification 3.3.1

ITS

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
	b. Main Vent Stack				See 3.3.3
	High Range	D	R	Q	
	Mid Range	D	R	Q	
	c. Spent Fuel Pit Lower Level				MZ1
	High Range	D	R	Q	
[T3.3.1-1(14)] 39.	Steam/Feedwater Flow Mismatch	~~N.A.~~ SR 3.3.1.1	~~R~~ SR 3.3.1.10	~~M~~ SR 3.3.1.7	L16
[T3.3.1-1(14)] 40.	Low Steam Generator Water Level	~~N.A.~~ SR 3.3.1.1	~~R~~ SR 3.3.1.10	~~M~~ SR 3.3.1.7	
41.	CV Level (Wide Range)+	M	R	R	
42.	CV Pressure (Wide Range)++	M	R	R	See 3.3.3
43.	CV Hydrogen Monitor+++	M	R	R	
44.	CV High Range Radiation Monitor++++	M	R#	R	LA4
45.	RCS High Point Vents	N.A.	N.A.	R	A7
[T3.3.1-1(1)] 46.	Manual Reactor Trip	N.A.	N.A.	~~R(1)~~ SR 3.3.1.14	~~(1) The manual reactor trip operational test shall verify the independent operability of the manual shunt trip circuit and the manual UV trip circuit on the reactor trip breakers. The test shall also verify the operability of the UV trip circuit on the bypass breakers.~~
[T3.3.1-1(1,18,19)] 47.	Reactor Trip Bypass Breakers	N.A.	N.A.	~~M(3), R(4)~~ SR 3.3.1.4 SR 3.3.1.14	~~(3) Remote manual UV trip required only when placing the bypass breaker in service. (4) Perform UV trip from protection system.~~ L20

Specification 3.3.1

A1

A1

ITS

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to plant operational safety instrumentation systems.

Objective

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification

[LCO 3.3.2] 3.5.1.1 The Engineered Safety Features initiation instrumentation ~~setting limits shall be as stated~~ in Table ~~3.5-5~~ 3.3.2-1 shall be OPERABLE.

[ACTION A] 3.5.1.2 For on-line testing or in the event of a subsystem instrumentation channel failure, plant operation at rated power shall be permitted to continue in accordance with Tables ~~3.5-2 through 3.5-5~~ 3.3.2-1.

3.5.1.3 In the event the number of channels in service listed in Table 3.5-5 falls below the limits given in the column entitled Minimum Channels Operable, operation shall be limited according to the requirement shown in Column 2. See 3.3.3

3.5.1.4 The containment ventilation isolation function is only required when containment integrity is required. See 3.3.6

[ACTION A] 3.5.1.5 In the event the number of operable channels of a particular functional unit listed in Tables ~~3.5-2, 3 or 4~~ 3.3.2-1 falls below the limits given in the column entitled Total Number of Channels, operation shall be limited according to the requirement shown in Column 3 immediately. M2

Amendment No. ~~85~~

License Amendment Request, 12/10/95

Specification 3.3.2

A1

TABLE 3.5-3 (Continued)

ITS

ENGINEERED SAFETY FEATURES INSTRUMENTATION LIMITING OPERATING CONDITIONS

TABLE NOTATIONS

[T3.3.2-1(Note A)] # Above Low Pressure SI Block Permit interlock

[T3.3.2-1(Note C)] ## Trip function may be blocked below Low T_{avg} Interlock setpoint

The reactor may remain critical below the Power Operating conditions with this feature inhibited for the purpose of starting reactor coolant pumps

See 3.3.5

[ACTION B] ACTION 11 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least the Hot Shutdown Condition within the next ~~8~~ 6 hours and the Cold Shutdown Condition within the following 30 hours.

M22

[ACTION C] [ACTION D] [ACTION E] [ACTION G] ACTION 12 With the number of OPERABLE channels one less than the Total Number of Channels, Power Operation may proceed ~~until performance of the next required operational test~~ provided the inoperable channel is placed into the tripped condition within ~~1~~ 6 hour.

L21

MODE 3 in 7 hrs, MODE 4 in 13 hrs, MODE 5 in 37 hrs

[ACTION I] ACTION 13 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 1 hour or be in ~~at least the Hot Shutdown Condition within the next 8 hours and the Cold Shutdown Condition within the following 30 hours.~~

M24

ACTION 14 With the number of OPERABLE channels one less than the Total Number of Channels; place the inoperable channel into the blocked condition within 1 hour, and restore the inoperable channel to OPERABLE status within 48 hours or be in at least the Hot Shutdown Condition within the next 8 hours and the Cold Shutdown Condition within the following 30 hours.

See 3.3.5

[ACTION C] or be in MODE 3 in 12 hours and MODE 5 in 42 hours

[ACTION D, G] or be in MODE 3 in 12 hours and MODE 4 in 18 hours

[ACTION E] or be in MODE 3 in 12 hours, MODE 4 in 18 hours, and MODE 5 in 36 hours

M23

Add RA E.1 "NOTE"

Add ACTIONS "Note"

A5

~~Amendment No.~~

License Amendment Request, 12/10/95

Specification 3.3.2

A1

TABLE 3.5-4 (Continued)

CTS

ISOLATION FUNCTIONS INSTRUMENTATION LIMITING OPERATING CONDITIONS

TABLE NOTATIONS

ACTION 15 With less than the Total Number of Channels, Power Operation may continue provided the Containment Ventilation Purge and Exhaust valves are maintained closed.

See 3.3.6

[ACTION F]

ACTION 16 With the number of channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.4.3.

M25

[ACTION B] be in MODE 3 in 54 hrs and MODE 5 in 84 hrs

[ACTION F] be in MODE 3 in 54 hrs and MODE 4 in 60 hrs

Amendment No.

License Amendment Request, 12/10/95

Specification 3.3.2

A1

ITS

TABLE 3.5-1

ENGINEERED SAFETY FEATURE SYSTEM INITIATION INSTRUMENT SETTING LIMITS

	NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
[T3.3.2-1(1.c)]	1.	High Containment Pressure (HI Level)	Safety Injection*	≤ ~~5~~ 4 psig
[T3.3.2-1(2.c)]	2.	High Containment Pressure (HI-HI Level)	a. Containment Spray** b. Steam Line Isolation	≤ ~~25~~ 20 psig
[T3.3.2-1(1.d)]	3.	Pressurizer Low Pressure	Safety Injection*	≥ ~~1700~~ 1715 psig
[T3.3.2-1(1.e)]	4.	High Differential Pressure Between any Steam Line and the Steam Line Header	Safety Injection*	≤ ~~150~~ 100 psi
[T3.3.2-1(4.d,4.e)]	5.	High Steam Flow in 2/3 Steam Lines***	a. Safety Injection* b. Steam Line Isolation	≤ ~~40~~% (at zero load) of full steam flow ≤ ~~40~~% (at 20% load) of full steam flow ≤ ~~110~~% (at full load) of full steam flow
		Coincident with Low T_{avg} or Low Steam Line Pressure		≥ ~~541~~ 543 °F T_{avg} ≥ ~~600~~ 614 psig steam line pressure
	6.	Loss of Power		
		a. 480V Emerg. Bus Undervoltage (Loss of Voltage) Time Delay	Trip Normal Supply Breaker	328 Volts ± 10% ≤ 1 sec when voltage is reduced to zero

M1

M1

See 3.3.5

Add SR "Note" A5

ITS

TABLE 4.1-1 (Continued)

	Channel Description	Check	Calibrate	Test	Remarks
21.	Containment Sump Level	N.A.	R	N.A.	See 3.4.15
22.	Turbine Trip Logic**	N.A.	N.A.	R	See 3.3.1
23.	Accumulator Level and Pressure	S	R	N.A.	
24.	Steam Generator Pressure	S	R	M	
25.	Turbine First Stage Pressure	S	R	M	
26.	DELETED				
[SR 3.3.2.2] 27.	Logic Channel Testing	N.A.	N.A.	M(1) S/U(2)	(1) Applicability: SI, Cont Spray: MODES 1,2,3; Stm Line Isol: MODES 1,2,3,4. During hot shutdown and power operations. ~~When periods of reactor cold shutdown and refueling extend this interval beyond one month, this test shall be performed prior to startup.~~ M19 (2) Logic channel testing for nuclear source range channels shall only be required prior to each reactor startup, if not performed within the previous seven (7) days. See 3.3.1 L18
28.	DELETED				
29.	4 Kv Frequency	N.A.	R	R	See 3.3.1

on a STAGGERED TEST BASIS

** ~~Stop valve closure or low EH fluid pressure.~~

Specification 3.3.2

A1

Amendment No. ~~83~~, ~~106~~, ~~122~~, ~~127~~, 133

ITS

Specification 3.3.2

A1

4.5 EMERGENCY CORE COOLING, CONTAINMENT COOLING AND IODINE REMOVAL SYSTEMS TESTS

Applicability

Applies to testing of the Emergency Core Cooling and the Containment Spray (Iodine Removal) Systems.

Objective

To verify that the subject systems will respond promptly and perform their design functions, if required.

Specification

4.5.1 System Tests

Safety Injection System

[SR 3.3.2.8] 4.5.1.1 System tests shall be performed at each reactor refueling interval. The test shall be initiated by a test safety injection signal and shall be performed in such a manner as to prevent the injection of safety injection system fluid into the reactor coolant system.

18 months

M26

actual or simulated

A9

4.5.1.2 The test will be considered satisfactory if control board indication and visual observation indicate that all components have received the safety injection signal in the proper sequence and timing; i.e., the appropriate pump breakers shall have closed, and all automatic valves shall have completed their travel.

See 3.5.2

ITS

Specification 3.3.2

Containment Spray System

18 MONTHS M26 A1

4.5.1.3 System tests shall be performed at each refueling interval. The test shall be performed with the isolation valves in the spray supply lines at the containment and spray additive tank blocked closed. Operation of the system is initiated by tripping the ~~[illegible]~~ actuation instrumentation.

[3.3.2.8]

see 3.6.6 + 3.6.7

actual or simulated A9

4.5.1.4 Verify each spray nozzle is unobstructed at least every 10 years.

4.5.1.5 The tests discussed in 4.5.1.3 and 4.5.1.4 will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

See 3.6.6 + 3.6.7

Containment Fan Coolers

4.5.1.6 Each fan cooler unit shall be tested at monthly intervals to verify proper operation of all essential features including valves, dampers and piping.

4.5.2 Component Verification

4.5.2.1 When the reactor coolant pressure is in excess of 1,000 psi, it shall be verified at least once per 12 hours (from the RTGB indicators/controls) that the following valves are in their proper position with control power to the valve operators removed.

See 3.5.2

Valve Number	Valve Position
1- MOV 862 A&B	Open
2- MOV 863 A&B	Closed
3- MOV 864 A&B	Open
4- MOV 866 A&B	Closed

Specification 3.3.2

A1

TABLE 4.1-3

FREQUENCIES FOR EQUIPMENT TESTS

		Check	Frequency	Maximum Time Between Tests	
1.	Control Rods	Rod drop times of all full length rods	Each refueling shutdown	NA	See 3.1.4
2.	Control Rod	Partial movement of all full length rods	Every 2 weeks during reactor critical operations	20 days	
3.	Pressurizer Safety Valves	Set point	Each refueling shutdown	NA	See 3.4.10
4.	Main Steam Safety Valves	Verify each required MSSV lift setpoint per Table 4.1-4 in accordance with the Inservice Testing Program. Following testing, lift setting shall be within +/- 1%	In accordance with the Inservice Testing Program	NA	See 3.7.1
5.	Containment Isolation Trip	~~Functioning~~ TADOT	~~Each refueling shutdown~~ 18 months	NA	
6.	Refueling System Interlocks	Functioning	Prior to each refueling shutdown	NA	See 3.9.1
7.	Service Water System	Functioning	Each refueling shutdown	NA	see 3.7.7
8.	~~DELETED~~				
9.	Primary System Leakage	Evaluate	Daily when reactor coolant system is above cold shutdown condition	NA	See 3.4.13
10.	Diesel Fuel Supply	Fuel Inventory	Weekly	10 days	see 3.8.3
11.	~~DELETED~~				
12.	Turbine Steam Stop, Control, Reheat Stop, and Interceptor Valves	Closure	Quarterly during power operation and prior to startup	115 days	see 3.7.1

[TABLE 3.3.2-1(3)] [SR 3.3.2.6]

Add SR 3.3.2.6 "Note"

A10

LAR dated 1/29/96

~~Amendment No. 142, 155, 164~~

Specification 3.3.2

A1

TABLE 3.5-3

ITS

ENGINEERED SAFETY FEATURES INSTRUMENTATION LIMITING OPERATING CONDITIONS

A27

	NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 ~~MINIMUM CHANNELS OPERABLE~~	3 OPERATOR ACTION IF COLUMN 1 OR 2 CANNOT BE MET	APPLICABLE CONDITIONS
	1.	SAFETY INJECTION				
[T3.3.2-1(1.a)]		A. Manual	2	2	ACTION ~~11~~ B	MODE 1,2,3,4 >200°F
[T3.3.2-1(1.c)]		B. High Containment Pressure (Hi Level)	3	2	ACTION ~~12~~ E	>200°F
[T3.3.2-1(1.e)]		C. High Differential Pressure between Any Steam Line and the Steam Header	3/Steam Line	~~2/Steam Line~~	ACTION ~~12~~ D	MODES 1,2,3 (a) #
[T3.3.2-1(1.d)]		D. Pressurizer Low Pressure	3	2	ACTION ~~12~~ D	MODES 1,2,3 (a) # (c) A11
[T3.3.2-1(4.d)]		E. High Steam Flow In 2/3 Steam Lines Coincident with Low T_{avg} in 2/3 loops.	2/Steam Line and 1 T_{avg}/ Loop	~~1/Steam Line and 1 T_{avg} in 2 Loops OR 2/Steam Line and 1 T_{avg}~~	ACTION ~~12~~ D	MODES 1,2 (b),3 (b) ≥350 °F ## (c)
[T3.3.2-1(4.e)]		F. High Steam Flow In 2/3 Steam Lines Coincident with Low Steam Pressure in 2/3 lines	2/Steam Line and 1 Press/ Line	~~1/Steam Line and 1 Press in 2 Lines OR 2/Steam Line and 1 Press~~	ACTION ~~12~~ D	MODES 1,2 (b),3 (b) ≥350 °F ##
	2.	CONTAINMENT SPRAY				
[T3.3.2-1(2.a)]		A. Manual	2	2	ACTION ~~13~~ I	MODES 1,2,3,4 >200 °F
[T3.3.2-1(2.c)]		B. High Containment Pressure (Hi Hi Level)	3/Set	2/Set	ACTION ~~12~~ E	>200 °F

M24

~~Amendment No. 85~~

License Amendment Request, 12/10/95

Specification 3.3.2

A1

TABLE 3.5-4

ITS

ISOLATION FUNCTIONS INSTRUMENTATION LIMITING OPERATING CONDITIONS

A27

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 MINIMUM CHANNELS OPERABLE	3 OPERATOR ACTION IF COLUMN 1 OR 2 CANNOT BE MET	APPLICABLE CONDITIONS
1.	CONTAINMENT ISOLATION				
	A. Phase A				
[T3.3.2-1(3.a.3)]	i. Safety Injection	See Item No. 1 of Table 3.5-3 for all Safety Injection initiating functions and requirements			
[T3.3.2-1(3.a.1)]	ii. Manual	2	2	ACTION ~~N~~ B	MODES 1,2,3,4 >200 °F
[T3.3.2-1(3.b)]	B. Phase B	See Item No. 2 of Table 3.5-3 for all Containment Spray initiating functions and requirements			
	C. Ventilation Isolation				
	i. High Containment Activity, Gaseous	1	0	ACTION 15	During Containment Purge
	ii. High Containment Activity, Particulate	1	0	ACTION 15	During Containment Purge
	iii. Phase A	See Item No. 1.A of Table 3.5-4 for all Phase A initiating functions and requirements			

A27

A27

See 3.3.6

Add Table 3.3.2-1 Items 1.b, 2.b, 3.a.(2), 3.b(2), 4.b and 5.a

Add Actions C, G

L13

Amendment No. ~~85~~

License Amendment Request, 12/10/95

A1

TABLE 3.5-4 (Continued)

ITS

ISOLATION FUNCTIONS INSTRUMENTATION LIMITING OPERATING CONDITIONS

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 MINIMUM CHANNELS OPERABLE	3 OPERATOR ACTION IF COLUMN 1 OR 2 CANNOT BE MET	APPLICABLE CONDITIONS
2.	STEAM LINE ISOLATION				
[T3.3.2-1(1.f)]	A. High Steam Flow in 2/3 Steam Lines Coincident with Low T_{avg} in 2/3 loops	See Item No. 1.E of Table 3.5-3 for initiating functions and requirements			
[T3.3.2-1(1.g)]	B. High Steam Flow in 2/3 Steam Lines Coincident with Low Steam Pressure in 2/3 lines	See Item No. 1.F of Table 3.5-3 for initiating functions and requirements			
[T3.3.2-1(4.c)]	C. High Containment Pressure (Hi Hi Level)	See Item No. 2.B of Table 3.5-3 for initiating functions and requirements			
[T3.3.2-1(4.a)]	D. Manual	1/Line	~~1/Line~~	ACTION ~~16~~ F	≥350 °F MODES 1,2,3 (b) (b)
3.	FEEDWATER LINE ISOLATION				
[T3.3.2-1(5)]	A. Safety Injection	See Item No. 1 of Table 3.5-3 for all Safety Injection initiating functions and requirements			

A27

M25

Add: ACTION H

SR 3.3.2.1 SR 3.3.2.5

SR 3.3.2.3 SR 3.3.2.7

SR 3.3.2.4

T3.3.2-1 Item 6

M27

Add T3.3.2-1 "Allowable Value" column

M12

Amendment No. 85

License Amendment Request, 12/10/95

ITS

A1

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to plant operational safety instrumentation systems.

Objective

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification

3.5.1.1 The Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3.5-1. — See 3.3.2

[LCO 3.3.3] 3.5.1.2 For on-line testing or in the event of a subsystem instrumentation channel failure, plant operation ~~at rated power~~ in MODES 1, 2 and 3 shall be permitted to continue in accordance with Tables ~~3.5-2 through 3.5-5~~ 3.3.3-1. — M28

[LCO 3.3.3] 3.5.1.3 In the event the number of channels in service listed in Table ~~3.5-5~~ 3.3.3-1 falls below the limits given in the column entitled ~~Minimum~~ Required Channels ~~Operable~~, operation shall be limited according to the requirement shown in Column ~~2~~ Condition.

3.5.1.4 The containment ventilation isolation function is only required when containment integrity is required. — See 3.3.6

3.5.1.5 In the event the number of operable channels of a particular functional unit listed in Tables 3.5-2, 3, or 4 falls below the limits given in the column entitled Total Number of Channels, operation shall be limited according to the requirement shown in Column 3. — See 3.3.1 3.3.2

Add ACTIONS "NOTE 1" — M29

~~Amendment No. 85~~

Add ACTIONS "NOTE 2" — A5

License Amendment Request, 12/10/95

Specification 3.3.3

A1

TABLE 3.5-5
(THIS TABLE APPLIES ~~WHEN THE RCS IS > 350°F~~ IN MODES 1, 2, 3)
INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT

ITS [T 3.3.3-1 ITEM #]	NO.	INSTRUMENT	1 MINIMUM CHANNELS OPERABLE	2 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 CANNOT BE MET
[12]	1	Pressurizer Level	2	~~See Item 9 Table 3.5-2~~
[19]	2	Auxiliary Feedwater Flow Indication (Primary Indication)	A,B,C,G	A,B,C,H ~~Note 1~~
		SD AFW Pump	1 per S/G	
		MD AFW Pump	1 per S/G	
	3	Reactor Coolant System Subcooling Monitor	1	Note 2
[22]	4	PORV Position Indicator (Primary)	1	~~Note 3~~
[23]	5	PORV Blocking Valve Position Indicator (Primary)	1	A,B,C,H ~~Note 3~~
[24]	6	Safety Valve Position Indicator (Primary)	1	~~Note 3~~
	7	Noble Gas Effluent Monitors*****		
		a. Main Steam Line	1 per steamline	Note 4
		b. Main Vent Stack		
		High Range	1	Note 4
		Mid Range	1	Note 4
		c. Spent Fuel Pit-Lower Level		
		High Range	1	Note 4
[10]	8	CV High Range Radiation Monitor****	2	~~Note 4~~
[7]	9	CV Level (Wide Range)*	2	A,B,C,H ~~Note 5~~
[8]	10	CV Pressure (Wide Range)**	2	~~Note 5~~
[11]	11	CV Hydrogen Monitor***	1	~~Note 6~~
	12	Reactor Vessel Level Instrumentation System (RVLIS)	2	Note 7
[15-18]	13	Incore Thermocouple (T/C)	2 T/C per core quadrant	A,B,C,G ~~Note 8~~

R1

M30

L22

M47

M31

G

3

R1

* Containment Water Level Monitor - NUREG-0737 Item II.F.1.5
** Containment Pressure Monitor - NUREG-0737 Item II.F.1.4
*** Containment Hydrogen Monitor - NUREG-0737 Item II.F.1.6
**** Containment High-Range Radiation Monitor - NUREG-0737 Item II.F.1.3
***** Noble Gas Effluent Monitors - NUREG-0737 Item II.F.1.1

A19

Specification 3.3.3

A1

ITS

TABLE 3.5-5 (Continued)

INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT

TABLE NOTATION

Note 1: The three AFW lines from the MD AFW pumps and three AFW lines from the SD AFW pump each contain one primary flow indicator (2 AFW flow paths per steam generator for a total of 6 AFW lines). These primary indicators are backed up by the narrow range steam generator level indications. If one ~~or more~~ of the direct AFW flow indicators becomes inoperable when the RCS is > 350°F, restore the indicator(s) to an operable status within ~~7~~ 30 days, or prepare and submit a special report to the NRC within the following 14 days detailing the cause(s) of the inoperable indicator(s), the actions being taken to restore the indicator(s) to an operating status, the estimated date for completion of the repairs, and any compensatory action being taken while the indicator(s) is operable. The action required when any of the back-up indicators of AFW flow are inoperable is described in Table 3.5-2.

A24 L23 See 5.6.6 A24

[ACTION A] [ACTION B]

initiate action in accordance with 5.6.6

Note 2: If both channels of the RCS subcooling monitor become inoperable when the RCS is > 350°F, restore at least one channel to an operable status within 7 days, or prepare and submit a special report to the NRC within the following 14 days detailing the cause(s) of the inoperable channels, the actions being taken to restore at least one channel to an operable status, the estimated date for completion of the repairs, and any compensatory action being taken while both channels are inoperable.

R1

Note 3: The Pzr PORVs and Pzr PORV blocking valves both incorporate limit switches for the direct (primary) means of position indication. The back-up method of position indication consists of the PRT pressure and a temperature element in a common line downstream of the valves. The Pzr safety relief valves incorporate a vibration monitoring system as the primary method of valve position indication. The back-up method of position indication consists of a temperature element downstream of each valve and PRT pressure. If the primary method of position indication for either the Pzr PORVs, Pzr PORV blocking valves, or Pzr safety relief valves becomes inoperable when the RCS is > 350°F, restore the primary method to an operable status within 7 days, or prepare and submit a special report to the NRC within the following 14 days detailing the cause of the inoperable primary position indication method, the actions being taken to restore it to an operable status, the estimated date for completion of the repairs, and any compensatory action being taken while the primary position indication method is inoperable. If any of the back-up methods of position indication for these valves becomes inoperable, it is to be repaired as soon as plant conditions permit.

A24

[ACTION D] [ACTION H]

initiate action in accordance with 5.6.6

See 5.6.6 A24

Add ACTION F A12

A1

TABLE 3.5-5 (Continued)

INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT

ITS

TABLE NOTATION

Note 1: The three AFW lines from the MD AFW pumps and three AFW lines from the SD AFW pump each contain one primary flow indicator (2 AFW flow paths per steam generator for a total of 6 AFW lines). These primary indicators are backed up by the narrow range steam generator level indications. If one or more of the direct AFW flow indicators becomes inoperable when the RCS is > 350°F, restore the indicator(s) to an operable status within 7 days, or prepare and submit a special report to the NRC within the following 14 days detailing the cause(s) of the inoperable indicator(s), the actions being taken to restore the indicator(s) to an operating status, the estimated date for completion of the repairs, and any compensatory action being taken while the indicator(s) is operable. The action required when any of the back-up indicators of AFW flow are inoperable is described in Table 3.5-2.

A24

[ACTION C]
[ACTION H]

two

Initiate action in accordance with 5.6.6

See 5.6.6

A24

Note 2: If both channels of the RCS subcooling monitor become inoperable when the RCS is > 350°F, restore at least one channel to an operable status within 7 days, or prepare and submit a special report to the NRC within the following 14 days detailing the cause(s) of the inoperable channels, the actions being taken to restore at least one channel to an operable status, the estimated date for completion of the repairs, and any compensatory action being taken while both channels are inoperable.

R1

Note 3: The Pzr PORVs and Pzr PORV blocking valves both incorporate limit switches for the direct (primary) means of position indication. The back-up method of position indication consists of the PRT pressure and a temperature element in a common line downstream of the valves. The Pzr safety relief valves incorporate a vibration monitoring system as the primary method of valve position indication. The back-up method of position indication consists of a temperature element downstream of each valve and PRT pressure. If the primary method of position indication for either the Pzr PORVs, Pzr PORV blocking valves, or Pzr safety relief valves becomes inoperable when the RCS is > 350°F, restore the primary method to an operable status within 7 days, or prepare and submit a special report to the NRC within the following 14 days detailing the cause of the inoperable primary position indication method, the actions being taken to restore it to an operable status, the estimated date for completion of the repairs, and any compensatory action being taken while the primary position indication method is inoperable. If any of the back-up methods of position indication for these valves becomes inoperable, it is to be repaired as soon as plant conditions permit.

A24

[ACTION D]
[ACTION H]

Initiate action in accordance with 5.6.6

See 5.6.6

A24

A1

ITS

TABLE 3.5-5 (Continued)

INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT

TABLE NOTATION

~~Note 4: With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, restore the inoperable Channel(s) to OPERABLE status within 7 days or, prepare and submit a Special Report to the NRC within the following 14 days detailing the cause of the inoperable Channel(s), the action being taken to restore the Channel(s) to operable status, the estimated date for completion of repairs, and any compensatory action being taken while the Channel(s) is inoperable.~~ R1

[ACTION A] [ACTION B] Note 5: If one channel is inoperable, restore the channel to operable status within 30 days or, prepare and submit a special report to the NRC within the following 14 days detailing the cause(s) of the inoperable channels, the actions being taken to restore the channel to operable status, the estimated date for completion of the repairs, and the compensatory action being taken while the channel is inoperable. If both channels become inoperable and a pre-planned alternate method of monitoring is available, then restore at least one channel to operable status within 7 days or prepare and submit a special report to the NRC within the following 14 days detailing the cause(s) of the inoperable channels, the action being taken to restore at least one channel to operable status, the estimated date for completion of the repairs, and a description of the alternate method of monitoring the affected parameter while both channels are inoperable. If a pre-planned alternate method of monitoring the affected parameter is not available and implemented with both channels inoperable, then restore at least one channel to an operable status within 7 days or be in Hot Shutdown within 6 hours and ≤ 350°F within the following 30 hours.

Initiate action per 5.6.6.

See 5.6.6 LA6 See 5.6.6 A13

[ACTION C] [ACTION H]

[ACTION E] [ACTION G] Note 6: With both channels inoperable, restore at least one channel to an operable status within ~~14 days~~ or be in ~~Hot Shutdown~~ within 6 hours and ~~≤ 200°F~~ within the following ~~30~~ hours.

MODE 3 MODE 4 72 hours 6 L24 M31 L24

Add Functions & Requirements:

- SG Pressure
- Cont. Spray Additive Tank Level
- Cont. Isolation Valve Position Indication
- SG Level
- Power Range Neutron Flux
- Source Range Neutron Flux
- RCS Pressure
- RCS Hot Leg Temperature
- RCS Cold Leg Temperature
- RWST Level
- CST Level

M32

A1

TABLE 3.5-5 (Continued)

INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT

ITS

TABLE NOTATION

Note 7:* ~~With the number of OPERABLE channels one less than the MINIMUM CHANNELS OPERABLE requirement, restore the inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.~~

~~With the number of operable channels two less than the MINIMUM CHANNELS OPERABLE requirement, restore at least one channel to operable status within 48 hours, or be in at least HOT SHUTDOWN within the next 12 hours.~~

R1

[ACTION A] [ACTION B] Note 8: With the number of operable thermocouples one less than required by the MINIMUM CHANNELS OPERABLE requirements, restore the inoperable thermocouples to operable status within ~~X~~ 30 days ~~or be in at least HOT SHUTDOWN within the next 12 hours and < 350°F within the next 30 hours.~~ initiate 5.6.6 immediately

L25

[ACTION C] [ACTION G] With the number of operable thermocouples two less than the MINIMUM CHANNELS OPERABLE requirement, restore at least one thermocouple to OPERABLE status within ~~48 hours~~ 7 days ~~or be in at least HOT SHUTDOWN within the next 12 hours and < 350°F within the next 30 hours.~~ MODE 3-6hrs; MODE 4-12hrs.

L26

M33

~~For the remainder of Cycle 13 and Cycle 14, Note 7 above is superseded by the following:~~

~~With the number of OPERABLE channels one less than the MINIMUM CHANNELS OPERABLE requirement, restore the inoperable channel to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission within the following 14 days outlining the action taken, the cause of inoperability, and the plans and schedule for restoring the system to OPERABLE status.~~

~~With the number of operable channels two less than the MINIMUM CHANNELS OPERABLE requirement, ensure the availability of an alternate method of monitoring the reactor vessel inventory. Restore at least one channel to operable status within 48 hours and prepare and submit a Special Report to the Commission within the following 14 days outlining the action taken, the cause of inoperability, and the plans and schedule for restoring the system to OPERABLE status.~~

A14

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibration	Test	Remarks
32.	Loss of Power				See 3.3.5
	a. 480 Emerg. Bus Undervoltage (Loss of Voltage)	N.A.	R	R	
	b. 480 Emerg. Bus Undervoltage (Degraded Voltage)	N.A.	R	R	
33.	Auxiliary Feedwater Flow**** Indication	M [SR 3.3.3.1]	R [SR 3.3.3.2]	N.A.	
34.	Reactor Coolant System** Subcooling Monitor	M	R	N.A.	R1
35.	PORV Position Indicator***	N.A. [SR 3.3.3.1]	N.A. [SR 3.3.3.2]	R	M34
36.	PORV Blocking Valve*** Position Indicator	N.A. [SR 3.3.3.1]	N.A. [SR 3.3.3.2]	R	
37.	Safety Relief Valve Position*** Indicator	N.A. [SR 3.3.3.1]	N.A. [SR 3.3.3.2]	R	
38.	Noble Gas Effluent Monitors***** a. Main Steam Line	D	R	Q	R1

** Instrument for Detection of Inadequate Core Cooling - NUREG 0578 Item 2.1.3.b.
*** Direct Indication of Power Operated Relief Valve and Safety Valve Position - NUREG 0578 Item 2.1.3.a.
**** Auxiliary Feedwater Flow Indication to Steam Generator - NUREG 0578 Item 2.1.7.b.
***** Noble Gas Effluent Monitors - NUREG-0737 Item II.F.1.1.

A19

Specification 3.3.3

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
	b. Main Vent Stack				
	High Range	D	R	Q	
	Mid Range	D	R	Q	
	c. Spent Fuel Pit Lower Level				
	High Range	D	R	Q	
39.	Steam/Feedwater Flow Mismatch	N.A.	R	M	
40.	Low Steam Generator Water Level	N.A.	R	M	
41.	CV Level (Wide Range)+	M [SR 3.3.3.1]	R [SR 3.3.3.2]	R	
42.	CV Pressure (Wide Range)++	M [SR 3.3.3.1]	R [SR 3.3.3.2]	R	
43.	CV Hydrogen Monitor+++	M [SR 3.3.3.1]	R [SR 3.3.3.2]	R	
44.	CV High Range Radiation Monitor++++	M [SR 3.3.3.1]	R [SR 3.3.3.2]	R	
45.	RCS High Point Vents	N.A.	N.A.	R	
46.	Manual Reactor Trip	N.A.	N.A.	R(1)	(1) The manual reactor trip operational test shall verify the independent operability of the manual shunt trip circuit and the manual UV trip circuit on the reactor trip breakers. The test shall also verify the operability of the UV trip circuit on the bypass breakers.
47.	Reactor Trip Bypass Breakers	N.A.	N.A.	M(3),R(4)	(3) Remote manual UV trip required only when placing the bypass breaker in service. (4) Perform UV trip from protection system.

R1

See 3.3.1

L27

See 3.3.1

Specification 3.3.3

A1

Table 4.1-1 (Continued)

~~48.~~	~~Reactor Vessel Level Instrumentation System (RVLIS)~~	~~M~~	~~R~~	~~N.A~~
49.	Incore Thermocouple Temperature Instrumentation	M [SR 3.3.3.1]	R [SR 3.3.3.2]	N.A

R1

~~+ Containment Water Level Monitor - NUREG-0737 Item II.F.1.5~~
~~++ Containment Pressure Monitor - NUREG-0737 Item II.F.1.4~~
~~+++ Containment Hydrogen Monitor - NUREG-0737 Item II.F.1.6~~
~~++++ Containment High-Range Radiation Monitor - NUREG-0737 Item II.F.1.3~~
~~# CP&L's letter dated April 28, 1982, S. R. Zimmerman to S. A. Varga, provides an acceptable alternate calibration methodology.~~

A19

~~S~~	~~- At least once per 12 hours~~		~~Q~~	~~-~~	~~At least once per 92 days~~
~~D~~	~~- At least once per 24 hours~~		~~S/U~~	~~-~~	~~Prior to each reactor startup if not performed in the previous seven (7) days~~
~~W~~	~~- At least once per 7 days~~				
~~B/W~~	~~- At least once per 14 days~~		~~R~~	~~-~~	~~At least once per 18 months~~
~~M~~	~~- At least once per 31 days~~		~~N.A.~~	~~-~~	~~Not applicable~~

Add SR "NOTE" — A15

Add SR 3.3.3.2 "NOTE" — A16

Add ITS Specification 3.3.4 — M35

A1

Specification 3.3.3
3.3.4

A1

TABLE 3.5-3 (Continued)

ITS

ENGINEERED SAFETY FEATURES INSTRUMENTATION LIMITING OPERATING CONDITIONS

A27

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 ~~MINIMUM CHANNELS OPERABLE~~	3 OPERATOR ACTION IF COLUMN 1 OR 2 CANNOT BE MET	APPLICABLE CONDITIONS
3.	LOSS OF POWER				
[LCO 3.3.5]	A. 480V Emerg. Bus Undervoltage (Loss of Voltage)	2/Bus	~~1/Bus~~	ACTION 14	~~Reactor Critical~~
[LCO 3.3.5]	B. 480V Emerg. Bus Undervoltage (Degraded Voltage)	3/Bus	~~2/Bus~~	ACTION 14	~~Reactor Critical~~ ###

Shall be OPERABLE

MODES 1, 2, 3, and 4, when associated DG is required to be OPERABLE by LCO 3.8.2

M36

Add ACTIONS "NOTE"

A5

~~Amendment No. 87, 136~~

License Amendment Request 12/10/95

Specification 3.3.5

A1

TABLE 3.5-3 (Continued)

ITS

ENGINEERED SAFETY FEATURES INSTRUMENTATION LIMITING OPERATING CONDITIONS

TABLE NOTATIONS

Above Low Pressure SI Block Permit interlock

Trip function may be blocked below Low T_{avg} Interlock setpoint

See 3.3.2

The reactor may remain critical below the Power Operating conditions with this feature inhibited for the purpose of starting reactor coolant pumps

[Applicability NOTE]

ACTION 11 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least the Hot Shutdown Condition within the next 8 hours and the Cold Shutdown Condition within the following 30 hours.

ACTION 12 With the number of OPERABLE channels one less than the Total Number of Channels, Power Operation may proceed until performance of the next required operational test provided the inoperable channel is placed into the tripped condition within 1 hour.

ACTION 13 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 1 hour or be in at least the Hot Shutdown Condition within the next 8 hours and the Cold Shutdown Condition within the following 30 hours.

[ACTION A]
[ACTION D]

ACTION 14 With the number of OPERABLE channels (per bus) one less than the Total Number of (bypass) Channels; place the inoperable channel into the ~~blocked~~ condition within 6 ~~1~~ hour, ~~and restore the inoperable channel to OPERABLE status within 48 hours or be in at least the Hot Shutdown Condition within the next 8 hours and the Cold Shutdown Condition within the following 30 hours.~~ L28

One or more Loss of Voltage Functions

or enter applicable Condition(s) and RA(s) for the associated DG made inoperable by LOP DG start instrumentation immediately

~~Amendment No.~~

License Amendment Request, 12/10/95

Specification 3.3.5

A1

ITS

TABLE 3.5-3 (Continued)

ENGINEERED SAFETY FEATURES INSTRUMENTATION LIMITING OPERATING CONDITIONS

TABLE NOTATIONS

Above Low Pressure SI Block Permit interlock

Trip function may be blocked below Low T_{avg} Interlock setpoint

See 3.3.2

[Applicability NOTE] ### The reactor may remain critical below the Power Operating conditions with this feature inhibited for the purpose of starting reactor coolant pumps

ACTION 11 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least the Hot Shutdown Condition within the next 8 hours and the Cold Shutdown Condition within the following 30 hours.

ACTION 12 With the number of OPERABLE channels one less than the Total Number of Channels, Power Operation may proceed until performance of the next required operational test provided the inoperable channel is placed into the tripped condition within 1 hour.

ACTION 13 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 1 hour or be in at least the Hot Shutdown Condition within the next 8 hours and the Cold Shutdown Condition within the following 30 hours.

[ACTION B]
[ACTION D]

ACTION 14 With the number of OPERABLE channels per bus one less than the Total Number of Channels; place the inoperable channel into the ~~blocked~~ tripped condition within ~~1~~ 6 hour, and restore the inoperable channel to OPERABLE status within 48 hours or be in at least the Hot Shutdown Condition within the next 8 hours and the Cold Shutdown Condition within the following 30 hours. L29

One or more Degraded Voltage Functions

or enter applicable Condition(s) and RA(s) for the associated DG made inoperable by LOP DG start instrumentation immediately

Add RA B.1 "NOTE" L30

Add ACTION C M37

Amendment No.

License Amendment Request, 12/10/95

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibration	Test	Remarks
32.	Loss of Power				
	a. 480 Emerg. Bus Undervoltage (Loss of Voltage)	N.A.	(R) SR 3.3.5.2	(R) SR 3.3.5.1	
	b. 480 Emerg. Bus Undervoltage (Degraded Voltage)	N.A.	(R) SR 3.3.5.2	(R) SR 3.3.5.1	
33.	Auxiliary Feedwater Flow**** Indication	M	R	N.A.	
34.	Reactor Coolant System** Subcooling Monitor	M	R	N.A.	
35.	PORV Position Indicator***	N.A.	N.A.	R	
36.	PORV Blocking Valve*** Position Indicator	N.A.	N.A.	R	
37.	Safety Relief Valve Position*** Indicator	N.A.	N.A.	R	
38.	Noble Gas Effluent Monitors*****				
	a. Main Steam Line	D	R	Q	

See 3.3.3

** Instrument for Detection of Inadequate Core Cooling - NUREG 0578 Item 2.1.3.b.
*** Direct Indication of Power Operated Relief Valve and Safety Valve Position - NUREG 0578 Item 2.1.3.a.
**** Auxiliary Feedwater Flow Indication to Steam Generator - NUREG 0578 Item 2.1.7.b.
***** Noble Gas Effluent Monitors - NUREG-0737 Item II.F.1.1.

Specification 3.3.5

Specification 3.3.5

A1

ITS

TABLE 3.5-1

ENGINEERED SAFETY FEATURE SYSTEM INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
1.	High Containment Pressure (HI Level)	Safety Injection*	≤ 5 psig
2.	High Containment Pressure (HI-HI Level)	a. Containment Spray** b. Steam Line Isolation	≤ 25 psig
3.	Pressurizer Low Pressure	Safety Injection*	≥ 1700 psig
4.	High Differential Pressure Between any Steam Line and the Steam Line Header	Safety Injection*	≤ 150 psi
5.	High Steam Flow in 2/3 Steam Lines***	a. Safety Injection* b. Steam Line Isolation	≤ 40% (at zero load) of full steam flow ≤ 40% (at 20% load) of full steam flow ≤ 110% (at full load) of full steam flow
	Coincident with Low T_{avg} or Low Steam Line Pressure		≥ 541°F T_{avg} ≥ 600 psig steam line pressure
6.	Loss of Power		
SR 3.3.5.2.a	a. 480V Emerg. Bus Undervoltage (Loss of Voltage) Time Delay	Trip Normal Supply Breaker	328 Volts ± 10% ≤ 1 sec when voltage is reduced to zero

See 3.3.2

TABLE 3.5-1 (Continued)

ENGINEERED SAFETY FEATURE SYSTEM INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
[SR 3.3.5.2.b] 6. (Cont'd)	b. 480V Emerg. Bus Undervoltage (Degraded Voltage) Time Delay	Trip Normal Supply Breaker	430 Volts ± 4 Volts 10.0 Second Delay ± 0.5 sec.
7.	Containment Radioactivity High	Ventilation Isolation	The alarm is set with a method described in the ODCM.

See 3.3.6

~~• Initiates also containment isolation (Phase A), feedwater line isolation and starting of all containment fans.~~
~~•• Initiates also containment isolation (Phase B).~~
~~••• Derived from equivalent WP measurements.~~

A17

Specification 3.3.5

A1

A1

ITS

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to plant operational safety instrumentation systems.

Objective

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification

3.5.1.1 The Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3.5-1. (See 3.3.2)

3.5.1.2 For on-line testing or in the event of a subsystem instrumentation channel failure, plant operation at rated power shall be permitted to continue in accordance with Tables 3.5-2 through 3.5-5. (See 3.3.1, 3.3.2, 3.3.3)

3.5.1.3 In the event the number of channels in service listed in Table 3.5-5 falls below the limits given in the column entitled Minimum Channels Operable, operation shall be limited according to the requirement shown in Column 2. (See 3.3.3)

[LCO 3.3.6 Applicability]

3.5.1.4 The containment ventilation isolation ~~function is only required when containment integrity is required.~~ Instrumentation for each function in T3.3.6-1 shall be OPERABLE During CORE ALTERATIONS, movement of fuel in Containment (L34)

3.5.1.5 In the event the number of operable channels of a particular functional unit listed in Tables 3.5-2, 3, or 4 falls below the limits given in the column entitled Total Number of Channels, operation shall be limited according to the requirement shown in Column 3. (See 3.3.1, 3.3.2)

Amendment No. 85

License Amendment Request 12/10/95

A1

TABLE 3.5-4 (Continued)

ITS

ISOLATION FUNCTIONS INSTRUMENTATION LIMITING OPERATING CONDITIONS

TABLE NOTATIONS

[ACTION A.1] ACTION 15 - With one radiation monitoring ~~less than the Total Number of~~ Channels inoperable. Power Operation may continue provided the Containment Ventilation Purge and Exhaust valves are maintained closed.

ACTION 16 - With the number of channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.4.3.

See 3.3.2

Add ACTIONS "NOTE" — A5

Add ACTION A.2 — M38

Add SR "NOTE" — A5

Amendment No.

License Amendment Request 12/10/95

Specification 3.3.6

A1

TABLE 3.5-4

ISOLATION FUNCTIONS INSTRUMENTATION LIMITING OPERATING CONDITIONS

A27

ITS

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 ~~MINIMUM CHANNELS OPERABLE~~	3 OPERATOR ACTION IF COLUMN 1 OR 2 CANNOT BE MET	APPLICABLE CONDITIONS
1.	CONTAINMENT ISOLATION				
	A. Phase A				
	i. Safety Injection	See Item No. 1 of Table 3.5-3 for all Safety Injection initiating functions and requirements			
	ii. Manual	2	2	ACTION 11	>200 °F
	B. Phase B	See Item No. 2 of Table 3.5-3 for all Containment Spray initiating functions and requirements			
	C. Ventilation Isolation				
[T3.3.6-1(3.a)]	i. High Containment Activity, Gaseous	1	~~0~~	ACTION ~~15~~ A	During Containment Purge
[T3.3.6-1(3.b)]	ii. High Containment Activity, Particulate	1	~~0~~	ACTION ~~15~~ A	During Containment Purge
	iii. ~~Phase A~~	~~See Item No. 1.A of Table 3.5-4 for all Phase A initiating functions and requirements~~			

See 3.3.2

CORE ALTERATIONS and movement of irradiated fuel within containment

M40

A27

A23

Add Table 3.3.6-1 Functions 1 and 2

M41

~~Amendment No. 85~~

License Amendment Request 12/10/95

TABLE 3.5-1 (Continued)

ENGINEERED SAFETY FEATURE SYSTEM INITIATION INSTRUMENT SETTING LIMITS

ITS

NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
6. (Cont'd)	b. 480V Emerg. Bus Undervoltage (Degraded Voltage) Time Delay	Trip Normal Supply Breaker	430 Volts ± 4 Volts 10.0 Second Delay ± 0.5 sec.
[T3.3.6-1(3)] 7.	Containment Radioactivity High	Ventilation Isolation	The alarm is set with a method described in the ODCM. 2 x Background

See 3.3.5

LA5

M46

See 3.3.5

* Initiates also containment isolation (Phase A), feedwater line isolation and starting of all containment fans.

** Initiates also containment isolation (Phase B).

*** Derived from equivalent WP measurements.

Specification 3.3.6

A1

Specification 3.3.6
3.3.7
A1

3.8 REFUELING

Applicability

Applies to operating limitations during refueling operations.

Objective

To minimize the possibility of an accident occurring during refueling operations that could affect public health and safety.

Specification

3.8.1 During refueling operations the following conditions shall be satisfied:

a. The equipment door and at least one door in the personnel air lock shall be properly closed. For those systems which provide a direct path from containment atmosphere to the outside atmosphere, all automatic containment isolation valves shall be operable or at least one valve shall be securely closed in each line penetrating the containment. See 3.9.3

b. The containment vent and purge system, including the radiation monitors which initiate isolation shall be tested and verified to be operable immediately prior to refueling operations. L31

c. Radiation levels in the containment and spent fuel storage areas shall be monitored continuously. See 3.9.2 + 3.9.3

d. Whenever core geometry is being changed, core subcritical neutron flux shall be continuously monitored by at least two source range neutron monitors, each with continuous visual indication in the control room and one with audible See 3.9.2

Add SR 3.3.6.1
SR 3.3.6.2
SR 3.3.6.3 (and Note)
SR 3.3.6.4
M42

Add Specification 3.3.7
M43

Specification 3.3.8

ITS

3.4.6 [ACTION A]

A1

~~In the event that the number of channels of the Auxiliary Feedwater Initiation circuits falls below the limits given in the column entitled Minimum Operable Channels, or Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirements shown in Column 3 of Table 3.4-1. The Auxiliary Feedwater System Automatic Initiation Setting Limits are shown in Table 3.4-2. If the setpoint is less conservative than the value shown in the Allowable Values column to Table 3.4-2, declare the channel inoperable and operation shall be limited according to the requirement shown in Column 3 of Table 3.4-1.~~

one or more functions with one or more required channels or trains become inoperable, enter the condition referenced in Table 3.3.8-1 for the channel(s) or train(s), immediately

Add LCO 3.3.8 — A20

Add ACTIONS "NOTE" — A5

place channel in trip - 6 hrs, or MODE 3 - 12 hrs, and MODE 4 - 18 hrs

TABLE 3.4-1

AUXILIARY FEEDWATER FLOW AUTOMATIC INITATION*

ITS

	NO.	FUNCTIONAL UNIT	1 MINIMUM CHANNELS OPERABLE	2 MINIMUM DEGREE OF REDUNDANCY	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
[T3.3.8-1(1)] [ACTION B]	1.	Steam Gen. Water Level-low-low a. Start Motor-Driven Pumps b. Start Turbine-Driven Pump	2/Steam Generator 2/Steam Generator	1/Steam Generator 1/Steam Generator	Maintain Hot Shutdown Maintain Hot Shutdown (L32)
[T3.3.8-1(4)]	2.	Undervoltage-4KV Busses 1 & 4 Start Turbine-Driven Pump (15 Second Time Delay Pickup)	2 Per Bus	0	Note 1
[T3.3.8-1(2)]	3.	S.I. Start Motor-Driven Pumps	See Table 3.5-3, Item No.1		
[T3.3.8-1(3)]	4.	Station Blackout Start Motor-Driven Pumps (40 Second Time Delay Prior to Starting MD AFW Pumps on Blackout Sequence)	2 Per Bus	0	Note 2
[T3.3.8-1(5)]	5.	Trip of Main Feedwater Pumps Start Motor-Driven Pumps	1/Pump	0	Note 2 (A27)

[MODES 1,2,3] [MODES 1,2]

* This table is applicable whenever the RCS is > 350°F except Item 5. Item 5 is applicable only when the RCS is at normal operating temperature and the reactor is critical.

[ACTION B]

Note 1: ~~4KV Busses 1, 2, and 4 each have two undervoltage relays. One relay on each of the three busses provides an input to the reactor trip logic. Both relays on Busses 1 and 4 provide inputs to the SD AFW pump start logic. If the undervoltage relay on Busses 1 or 4 that provides the input to the reactor trip logic fails, follow the requirements of Table 3.5-2 Item 14 in addition to the following.~~ (A21) If either 4KV undervoltage relay on Busses 1 or 4 fails, within 6 hours (L33) insert the equivalent of an undervoltage signal from the affected relay in the SD AFW pump start circuit and (or) repair the affected relay within 7 days. If the affected relay is not repaired in the 7 days, then commence a normal plant shutdown to hot standby. MODE 3 in 12 hours, MODE 4 in 18 hours. (M48)

[ACTION D]

Note 2: Restore the inoperable channel to operable status within 48 hours. If the inoperable chanel is not restored to an operable status within 48 hours, then commence a normal plant shutdown and cooldown to ≤ 350°F (M44)

[ACTION C]

MODE 3 in 54 hrs.

MODE 4 in 60 hrs.

Specification 3.3.8 (A)

A1

TABLE 3.4-2

AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION SETPOINTS

ITS	FUNCTIONAL UNIT	SETTING LIMIT
	AUXILIARY FEEDWATER	
[T3.3.8-1(1)]	a. Steam Generator Water Level-low-low	≥ ~~14~~ 16% of narrow range instrument span each steam generator (M1)
[T3.3.8-1(4)]	b. Undervoltage - 4KV Busses 1 & 4	≥ ~~70~~ 3120% of 4KV Busses 1 & 4 Normal Voltage
[T3.3.8-1(2)]	c. S.I.	See Table ~~3.5-3, Item No. 1 and Table 3.5-1~~ 3.3.2-1, Function 1
[T3.3.8-1(3)]	d. Station Blackout	See ~~Table 3.5-1, Item No. 6~~ 328 V ± 10% with ≤ 1 sec. time delay

Add T 3.3.8-1 "Allowable Values" (M12)

Specification 3.3.8

CTS

[SR NOTE] 4.8.5 The surveillance requirements for auxiliary feedwater system automatic initiation shall be as stated in Table ~~4.8-1~~ 3.3.8-1. A1

Basis

The monthly testing of the auxiliary feedwater pumps by recirculation will verify their operability. The capacity of any one of three auxiliary feedwater pumps is sufficient to meet decay heat removal requirements.[1], [2], [3]

Proper functioning of the steam turbine admission valve and the starting of the feedwater pumps will demonstrate the integrity of the steam driven pump. Verification of correct operation will be made both from instrumentation within the control room and direct visual observation of the pumps. Testing of the steam turbine auxiliary feedwater pump is not required during periods of cold shutdown when steam is not available. In this condition the pump is not required for plant safety. If the steam turbine driven auxiliary feedwater pump has exceeded its surveillance interval following a period of reactor cold shutdown, then the surveillance test shall be performed prior to reactor criticality, but not to exceed 24 hours after the reactor coolant system has been at a minimum 547 deg. F. A4

References

(1) FSAR Section 10.4
(2) FSAR Section 14.1.11
(3) FSAR Section 14.2.5

ITS

TABLE 4.8-1

AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION
SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
	AUXILIARY FEEDWATER			
[T3.3.8-1(1)]	a. Steam Generator Water Level--Low-Low	N.A [SR3.3.8.1]	[SR3.3.8.4] See Table 4.1-1, Item 11	[SR3.3.8.2]
[T3.3.8-1(4)]	b. Undervoltage - 4 Kv busses 1 and 4	N.A.	R [SR3.3.8.4]	R [SR3.3.8.3]
[T3.3.8-1(2)]	c. S. I.	(all Safety Injection surveillance requirements)		
[T3.3.8-1(3)]	d. Station Blackout - E1 and E2 busses	N.A.	N.A [SR3.3.8.4]	R [SR3.3.8.3]
[T3.3.8-1(5)]	e. Trip of Main Feedwater Pumps	N.A.	N.A.	R [SR3.3.8.3]

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Specification 3.3.8

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.3 - INSTRUMENTATION

PART 2

DISCUSSION OF CHANGES (DOCS) FOR CTS MARKUP

ADMINISTRATIVE CHANGES

A1 In the conversion of the H. B. Robinson Steam Electric Plant (HBRSEP), Unit 2 Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS Specification 2.3.1.2.d describes in words, the amount by which the overtemperature ΔT trip setpoint is automatically reduced. ITS Specification 3.3.1 provides, instead, the mathematical expression for this reduction. This change is administrative, and has no adverse impact on safety.

A3 CTS Specification 2.3.1.2.f requires that the reactor be tripped on low reactor coolant flow. ITS Specification 3.3.1 has the same requirement, but identifies single loop low flow and two loop low flow separately. Since these numbers are identical, this change is administrative, and has no adverse impact on safety.

A4 The CTS Bases (and References) are not retained in the ITS, but are replaced in their entirety. The ITS includes significantly expanded and improved Bases. The Bases do not define or impose any specific requirements but serve to explain, clarify and document the reasons (i.e., Bases) for the associated Specification. The Bases are not part of the Technical Specifications required by 10 CFR 50.36. This change is administrative, and has no adverse impact on safety.

A5 The CTS is revised to adopt ISTS ACTIONS "Note," and/or Surveillance Requirements "Note." The ACTIONS "Note" provides for separate Condition entry for each function. The CTS is silent with regard to separate Condition entry, neither specifically permitting, nor disallowing. The Surveillance Requirements "Note" refers the reader to the specified ITS Table to determine which Surveillance Requirements apply for each Function. This change is administrative, and has no adverse impact on safety.

A6 CTS Table 3.5-2, Item 4.B is revised to add ITS footnote "e" to the Applicable Conditions of "Hot/Cold Shutdown." Footnote "e" basically notes that this Specification applies in those MODES when the reactor trip breakers are open. Item 4.C is specifically Applicable when the reactor trip breakers are closed. This change is therefore administrative, and has no adverse impact on safety.

A7 CTS Table 4.1-1 contains descriptive comments in the "Remarks" column. These remarks are not retained in the ITS. Deletion of these remarks does not result in any technical alteration of the Specifications. Therefore, this change is administrative and has no adverse impact on safety.

A8 The CTS is revised to adopt Surveillance Requirement "Notes," for SR 3.3.1.2 through SR 3.3.1.4, SR 3.3.1.6 through SR 3.3.1.12, SR 3.3.1.14, and SR 3.3.1.15. These NOTES provide clarifying instructions, such as when to adjust channels, when to perform the SR, what is included in the SR, and what is not included in the SR. This change is administrative, and has no adverse impact on safety.

A9 CTS Specifications 4.5.1.1 and 4.5.1.3 specify that system tests be initiated by "a test safety injection signal," or by "tripping the normal actuation instrumentation." ITS Surveillance Requirement SR 3.3.2.8 specifies that the test be initiated by an "actual or simulated actuation signal." This change is administrative, and has no adverse impact on safety.

A10 The CTS is revised to adopt the SR 3.3.2.6 "Note," which states that verification of setpoints is not required for manual initiation Functions. This change is administrative and has no adverse impact on safety.

A11 CTS Table 3.5-3, Functional Units "E," High Steam Flow coincident with Low T_{avg}, and "F," High Steam Flow coincident with Low Steam Pressure, have Applicability at a temperature $\geq$ 350°F. ITS Table 3.3.2-1 has Applicability for the same Functional Units in MODE 1, and in MODES 2 and 3, except when all Main Steam Isolation Valves (MSIVs) are closed. Since these Functions provide closure of the MSIVs under certain conditions, this change clarifies that the Functions are not needed whenever the MSIVs are already closed. This change is administrative, and has no adverse impact on safety.

A12 The CTS is revised to adopt ITS Specification 3.3.3 ACTION F. This Required Action is simply an instruction to enter the Condition referenced in Table 3.3.3-1 if other Required Actions and associated Completion Times are not met. This change is administrative, and has no adverse impact on safety.

A13 CTS Table 3.5-5 Note 5 is revised in the ITS to eliminate reference to a "pre-planned alternate method of monitoring" being available, and discussion of necessary actions if a pre-planned alternate method of monitoring is not available. This material is discussed in the Bases for ITS Specification 3.3.3, ACTION H, and is not necessary to be repeated in the ITS. This change is administrative, and has no adverse impact on safety.

A14 CTS Table 3.5-5, amended Note 7 is not adopted in the ITS. This amended Note was issued as a one-time requirement, which was applicable during the balance of Cycle 13 and Cycle 14. This change is administrative and has no adverse impact on safety.

A15 The CTS is revised by adopting the ITS Specification 3.3.3 Surveillance Requirement "Note," which merely states that SR 3.3.3.1 and SR 3.3.3.2 apply to each Post Accident Monitoring (PAM) Function in Table 3.3.3-1, except for Pressurizer Level, which is addressed in LCO 3.3.1. This change is administrative and has no adverse impact on safety.

A16 The CTS is revised to adopt the Note associated with SR 3.3.3.2, which states that Neutron Detectors are excluded from CHANNEL CALIBRATION. The Neutron Flux detectors are calibrated in a different manner, and separately from the channel. This change is administrative and has no adverse impact on safety.

A17 The CTS Table 3.5-1 footnotes are not retained in the ITS. These footnotes only provide descriptive information of a textbook nature related to specific Engineered Safety Feature (ESF) functional units, and need not be repeated in the ITS. This change is administrative, and has no adverse impact on safety.

A18 CTS Specification 3.10.5.1.a is revised to incorporate the phrase, "with undervoltage and shunt trip mechanisms," as part of Reactor Trip Breaker (RTB) OPERABILITY. Since the undervoltage and shunt trip mechanisms are considered to be a part of the RTB, this change only provides clarity, and is therefore administrative and has no adverse impact on safety.

A19 The footnotes in CTS Tables 3.5-5 and 4.1-1 are deleted. These footnotes only provide a reference to the source of a requirement in the Table, and need not be incorporated in the ITS. This change is administrative and has no adverse impact on safety.

A20 The CTS is revised by the addition of LCO 3.3.8, which simply makes the statement that the Auxiliary Feedwater (AFW) instrumentation in Table 3.3.8-1 must be OPERABLE. Therefore, this change is administrative and has no adverse impact on safety.

A21 CTS Table 3.4-1, Note 1, discussion of 4kV bus undervoltage relays and logic is not adopted in the ITS. This is descriptive information and direction of which requirements to follow if a relay fails. It is not necessary to repeat this information in the ITS, because the ITS is clear with respect to which requirements apply. This change is administrative and has no adverse impact on safety.

A22 CTS Table 4.8-1, Function "a" (Steam Generator Water Level - Low Low) and Function "d" (Station Blackout - E1 and E2 busses), are revised to require that a CHANNEL CHECK be performed at a Frequency of 12 hours,

and a CHANNEL CALIBRATION be performed at a Frequency of 18 months, respectively. Since CTS Table 4.1-1, Item 11 and Item 32.a already require that these Surveillances be performed at the same Frequencies, this change is administrative and has no adverse impact on safety.

A23 The CTS Table 3.5-4, item 1.C.iii reference to Phase A initiation of containment ventilation isolation is deleted. The Phase A containment ventilation isolation function is applicable in MODES 1, 2, 3, and 4, and the reference is not needed for ITS Specification 3.3.6, which has Applicability during CORE ALTERATIONS and during movement of irradiated fuel within containment. This change is administrative, and has no adverse impact on safety.

A24 Portions of the "Notes" in Table Notation to CTS Table 3.5-5 are not retained in the ITS. The portions that are deleted contain descriptive information related to the instrumentation in Table 3.5-5, and other clarification, which is not necessary to be repeated in the ITS. Since the deleted information does not contain any requirements, it is an administrative change which has no adverse impact on safety.

A25 CTS Table 4.1-1 is revised to add SRs 3.3.1.1, 3.3.1.2, 3.3.1,3, 3.3.1.4, 3.3.1.5, 3.3.1.6, 3.3.1.7, 3.3.1.8, 3.3.1.9, 3.3.1.10, 3.3.1.11, 3.3.1.12, 3.3.1.13, 3.3.1.14, and 3.3.1.15 into ITS. The addition of the surveillance text neither adds nor deletes requirements. The specific impact of the surveillance is discussed in each function for which the SR applies as shown in ITS Table 3.3.1-1. Therefore, this change is administrative, and has no impact on safety.

A26 CTS Table 3.5-2 is revised to add ITS Notes (c), (e), and (i) to Table 3.3.1-1. The specific impact of these notes is addressed with the Applicability of the Functions in which the Note is added in CTS Table 3.5-2. Therefore, this change is administrative, and has no impact on safety.

A27 CTS Table 3.5-2 is revised to delete Column 2, "Minimum Channels Operable," and the number of channels for Function 4.b in Column 1, "Total Number of Channels," is revised from two (2) to one (1). The Required Actions in the CTS refer to the total number of channels with few exceptions and the total number of channels is retained in the ITS Table 3.3.1-1. For the remaining Required Actions in the CTS that refer to the minimum channels OPERABLE, the minimum channels OPERABLE are the same number of channels as the total number of channels, with the exception of CTS Table 3.5-2, Function 4.b. The total number of channels for the source range with the reactor trip breakers open is changed from two (2) to (1), consistent with retention of CTS Required Action 5 in ITS as Required Action L.

CTS Tables 3.5-3 and 3.5-4 are revised to delete Column 2, "Minimum Channels Operable." The Required Actions in the CTS refer to the total

number of channels where the total number of channels differ with the minimum channels OPERABLE. The total number of channels is retained in the ITS Tables 3.3.2-1, 3.3.6-1, and in LCO 3.3.5.

CTS Table 3.4-1 is revised to delete Column 2, "Minimum Degree of Redundancy." The Required Actions in the CTS refer to the minimum channels OPERABLE. The total number of channels is retained in the ITS Tables 3.3.8-1.

This change neither adds or relaxes requirements. Therefore, this change is administrative, and has no impact on safety.

A28 CTS Table 3.5-2 Action 2 for Function 2.b, "Nuclear Flux Power Range Low Setpoint," is revised to ITS Required Action E. Action 2 provides requirements for the condition when THERMAL POWER is above 75% RTP, which is higher than the Nuclear Flux Power Range Low Setpoint, hence Action 2, Part b, could not be entered for an inoperability of the Nuclear Flux Power Range Low Setpoint. Required Action E is more appropriate for the Nuclear Flux Power Range Low Setpoint. Therefore, this change is administrative, and has no impact on safety.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 The CTS is revised to adopt the actual nominal trip setpoints that are used. These actual setpoints are more conservative than the CTS trip setpoint limits. The use of more conservative parameters is considered to be more restrictive, and has no adverse impact on safety.

M2 CTS Specification 3.5.1.5 and Table 3.5-2 ACTION 4 require that certain corrective actions be taken. ITS Specification 3.3.1 ACTIONS A and I, and ITS Specification 3.3.2 ACTION A, require that these corrective actions be taken "immediately." Since no time constraint currently exists, this change is more restrictive, and has no adverse impact on safety.

M3 CTS Table 3.5-2 ACTION 5 requires that compliance with shutdown margin be verified within 1 hour, and every 12 hours thereafter. ITS Specification 3.3.1 ACTION L requires, in addition, that activities involving positive reactivity addition be suspended immediately, and that unborated water source isolation valves be closed in 1 hour. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M4 CTS Table 3.5-2 Table Notation ACTION 6 permits operation to proceed, provided that the inoperable channel be placed in the tripped condition within 1 hour. ITS Specification 3.3.1 ACTION E requires instead, that the inoperable channel be placed in trip in 6 hours, or be in MODE 3 in 12 hours. This change imposes a shutdown requirement where such a requirement does not exist, and is therefore more restrictive and has no adverse impact on safety.

M5 CTS Table 3.5-2 ACTION 8 requires that an inoperable channel be restored to OPERABLE status within 48 hours, or open the reactor trip breakers. ITS Specification 3.3.1, ACTIONS C and K require that an inoperable channel be restored to OPERABLE status within 48 hours, or the reactor trip breakers be opened in 49 hours. This change imposes a time constraint that did not previously exist, which is therefore more restrictive and has no adverse impact on safety.

M6 CTS Table 3.5-2 ACTION 2 requires an inoperable channel be placed in trip within 1 hour, and <u>either</u>: a) power reduced to ≤ 75% RTP <u>and</u> power range flux trip setpoint reduced to ≤ 85% RTP in 4 hours <u>or</u>: b) QPTR be monitored every 12 hours. ITS Specification 3.3.1 ACTION D requires <u>either</u>: a) the inoperable channel be placed in trip within 6 hours <u>and</u> power reduced to ≤ 75% RTP in 12 hours, <u>or</u> b) the inoperable channel be placed in trip within 6 hours <u>and</u> SR 3.2.4.2 (QPTR) be performed once per 12 hours, <u>or</u> c) be in MODE 3 in 12 hours. The differences here are discussed from the perspective of the most and least restrictive actions that can be taken in response to the CONDITION of an inoperable power range neutron flux - high channel. The most restrictive actions that

can be taken in the CTS are to place the channel in trip in 1 hour, reduce THERMAL POWER to ≤ 75% RTP in 4 hours, and reduce the power range neutron flux trip setpoint to ≤ 85% RTP in 4 hours. The most restrictive action that can be taken in the ITS is to place the unit in MODE 3 in 12 hours. The action to shut down the unit is clearly a more restrictive change, and has no adverse impact on safety.

M7 CTS Table 3.5-2 ACTION 3 requires for an inoperable intermediate range neutron flux channel with THERMAL POWER above the P-6 setpoint, but below 10% RTP, that the inoperable channel be restored to OPERABLE status prior to increasing THERMAL POWER above 10% RTP. ITS Specification 3.3.1 ACTION F requires for an inoperable intermediate range neutron flux channel with THERMAL POWER above the P-6 setpoint, but below the P-10 setpoint, that THERMAL POWER either be reduced to below P-6 or increased above P-10 in 2 hours. The intermediate range neutron flux channels must be OPERABLE when the power level is above the capability of the source range and below the capability of the power range. The CTS has no time or action requirements for placing the unit in a condition where the power level is within the range of either the source range or power range instrumentation. The ITS requires decisive action be taken to place the unit in such a condition within a specified Completion Time. This change is more restrictive, and has no adverse impact on safety.

M8 CTS Specification 3.10.5.2 requires that, if an inoperable RTB or automatic trip logic train cannot be returned to OPERABLE status in 12 hours, the reactor be placed in the hot shutdown condition within the next 8 hours. ITS Specification 3.3.1 ACTION Q requires that an inoperable automatic trip logic train be restored to OPERABLE status in 6 hours, or be in MODE 3 in the next 6 hours. ACTION R requires that an inoperable RTB be restored to OPERABLE status in 1 hour, or be in MODE 3 in the next 6 hours. Since the ITS allowed outage times and Completion Times are shorter, this change is more restrictive, and has no adverse impact on safety.

M9 The CTS is revised to adopt ISTS Table 3.3.1-1 Items 18, 19 and 20 for Applicability in MODES 3, 4, and 5, including Required Actions C and V, with the RTBs closed. Since the CTS does not contain a similar Specification, this change is more restrictive, and has no adverse impact on safety.

M10 CTS Specification 3.10.5.3 requires that an inoperable RTB trip mechanism be restored to OPERABLE status in 48 hours or the unit be placed in the hot shutdown condition within the next 8 hour (56 hours total). ITS Specification 3.3.1 ACTION U requires that an inoperable RTB trip mechanism be restored to OPERABLE status in 48 hours or the unit be placed in MODE 3 in 54 hours and the RTB opened in 55 hours. Since the ITS Completion Times are smaller, this change is more restrictive, and has no adverse impact on safety.

M11 The CTS is revised to adopt ITS Specification 3.3.1 ACTIONS J, S, T, and V. Since no similar ACTIONS exist in the CTS for inoperable reactor trip instrumentation, this change is more restrictive, and has no adverse impact on safety.

M12 The CTS is revised to adopt the "ALLOWABLE VALUE" column in ITS Tables 3.3.1-1, 3.3.2-1, and 3.3.8-1. This column is added to provide an allowance for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those Reactor Protection System (RPS), Engineered Safety Features Actuation System (ESFAS), and AFW actuation channels that must function in harsh environments. The Allowable Values specified in these Tables conservatively set with respect to the analytical limits. The methodology used to calculate both the trip setpoints and allowable values is provided in the company setpoint methodology procedure. Since no similar Specifications for these instruments and functions exist in the CTS, this change is more restrictive and has no adverse impact on safety.

M13 The CTS is revised to adopt ITS Table 3.3.1-1 Functions (10) Reactor Coolant Pump (RCP) breaker position (single loop and two loops), (16) safety injection input from ESFAS, and RPS interlocks for intermediate range neutron flux, P-7, P-8, P-10, and turbine impulse pressure. Since no similar Specifications for these instruments and functions exist in the CTS, this change is more restrictive and has no adverse impact on safety.

M14 The CTS is revised to adopt ITS SR 3.3.1.3, which requires that results of incore detector measurements to NIS axial flux difference, and ITS SR 3.3.1.6, which requires calibration of the excore nuclear instrument channels to agree with incore detector measurements, for the OTΔT and OPΔT Functions. SR 3.3.1.1, SR 3.3.1.8, and SR 3.3.1.11 are adopted for the Power Range Neutron Flux-Low Function. SR 3.3.1.14 is adopted for the RCP Breaker Position and Safety Injection (SI) Input from ESFAS Functions. SR 3.3.1.11 and SR 3.3.1.13 are adopted for the RPS Interlock P-6, P-7, P-8, and P-10 Functions. Since no similar requirements exist in the CTS, this change is more restrictive and has no adverse impact on safety.

M15 CTS Table 4.1-1, Item 2 (Nuclear Intermediate Range) and Item 3 (Nuclear Source Range), require functional testing prior to each reactor startup if a functional test has not been performed in the previous 7 days. ITS SR 3.3.1.8 requires that a COT be performed prior to reactor startup, 4 hours after reducing power below P-10, 4 hours after reducing power below P-6, all if the COT has not been performed in the previous 92 days; and every 92 days thereafter. Since requirements similar to these do not exist in the CTS (with the exception of the requirement to

perform the COT prior to startup), this change is more restrictive and has no adverse impact on safety.

M16 CTS Table 4.1-1, Item 2 (Nuclear Intermediate Range) and Item 3 (Nuclear Source Range) are revised to adopt ITS Surveillance Requirement SR 3.3.1.11. Since no similar requirements exist in the CTS, this change is more restrictive, and has no adverse impact on safety.

M17 CTS Table 4.1-1, Item 22, Turbine Trip Logic, has Surveillance Requirements for only a test at refueling (R) intervals. ITS Surveillance Requirement SR 3.3.1.15 requires performance of a TADOT prior to reactor startup, when not performed in the previous 31 days. The CTS is also revised to adopt SR 3.3.1.10. No other similar requirement exists in the CTS. This change imposes more restrictive requirements, and has no adverse impact on safety.

M18 CTS Table 4.1-1, Item 27, Logic Channel Testing, requires monthly functional testing during hot shutdown and power operations, and for the source range channels prior to each reactor startup, if not performed within the previous 7 days. ITS SR 3.3.1.5 requires an ACTUATION LOGIC TEST be performed on a STAGGERED TEST BASIS, with Applicability in MODES 1 and 2; and in MODES 3, 4, and 5, when the RTBs are closed. Since this change imposes a broader Applicability, it is more restrictive and has no adverse impact on safety.

M19 CTS Table 4.1-1 requires logic channel testing be performed prior to startup, when periods of reactor cold shutdown and refueling extend the Surveillance interval beyond one month. ITS Surveillance Requirement SR 3.3.1.5 has Applicability in MODES 1 and 2; and MODES 3, 4, and 5 when the RTBs are closed. ITS SR 3.3.2.2 has Applicability in MODES 1, 2, and 3; and in one case, MODE 4. Since a Surveillance must be performed within its Frequency prior to entry into a MODE or other specified condition of Applicability, and the CTS requires performance of the SR prior to "startup," if it has not been performed within its Frequency, this change imposes more restrictive requirements, and has no adverse impact on safety.

M20 CTS Specification 2.3.1 is revised to add trip setpoints for ITS Table 3.3.1-1 Functions 3, 4, 14 and 15. The addition of specific setpoints to ITS is more restrictive, and this change has no adverse impact on safety.

M21 CTS Table 4.1-1, Item 39 (Steam/Feedwater Flow Mismatch) and Item 40 (Low Steam Generator Water Level) are revised to adopt ISTS SR 3.3.1.1, which requires that a CHANNEL CHECK be performed every 12 hours. Since no similar requirements exist in the CTS, this change is more restrictive and has no adverse impact on safety.

M22 CTS Table 3.5-3 ACTION 11, which applies to the manual SI actuation Function and the Manual containment Phase A isolation Function, requires the unit be in at least the Hot Shutdown condition within the next 8 hours. ITS Specification 3.3.2 ACTION B requires the unit be in MODE 3 within the next 6 hours. This change is more restrictive, and has no adverse impact on safety.

M23 CTS Table 3.5-3, ACTION 12 allows power operation to continue, provided the inoperable channel is placed in trip. ITS Specification Conditions C, D, E, and G contain the same provision. However, ITS Conditions D and G specify that, if the inoperable channel is not placed in trip within the allotted time, the unit must be in MODE 4 within specified Completion Times, and Conditions C and E specify the unit be placed in MODE 5 within specified Completion Times. While a shutdown is implied in the CTS, it is stated specifically in the ITS. This change imposes more restrictive requirements, and has no adverse impact on safety.

M24 CTS Table 3.5-3, Functional Unit 2.A, Manual Actuation of Containment Spray, has a required ACTION to restore the inoperable channel to OPERABLE status within 1 hour, or be in Hot Shutdown within the next 8 hours, and in Cold Shutdown in the following 30 hours. ITS Specification 3.3.2, ACTION I for the same Functional Unit, is to restore the inoperable channel to OPERABLE status in 1 hour, or be in MODE 3 in 7 hours, MODE 4 in 13 hours, and MODE 5 in 37 hours. This change imposes shorter Completion Time requirements, which is therefore more restrictive, and has no adverse impact on safety.

M25 CTS Table 3.5-4, Item 2.D, manual initiation of steam line isolation, requires in ACTION 16 that an inoperable channel be restored to OPERABLE status within 48 hours, or declare the associated valve inoperable and either restore it to OPERABLE status within the next 24 hours, or initiate procedures to place the unit in the hot shutdown condition. If the Specification is not met within an additional 48 hours, the reactor must be cooled to below 350°F. ITS Specification 3.3.2, Condition F requires the channel be restored to OPERABLE status within 48 hours, or be in MODE 3 within 54 hours, and in MODE 4 within 60 hours. This change imposes more restrictive requirements, and has no adverse impact on safety.

M26 CTS Specifications 4.5.1.1 and 4.5.1.3 require performance of system tests at each reactor refueling interval. The CTS does not explicitly limit the refueling interval to a finite time period. ITS Surveillance Requirement SR 3.3.2.8 requires performance of the test at an 18 month Frequency. This change is imposes more restrictive requirements, and has no adverse impact on safety.

M27 The CTS is revised to adopt ITS Specification 3.3.2 Condition H; Surveillance Requirements SR 3.3.2.1, SR 3.3.2.3 through SR 3.3.2.5, and

SR 3.3.2.7; and, Table 3.3.2-1 Item 6. Since no similar Specifications exist in the CTS, this change imposes new requirements and is therefore more restrictive and has no adverse impact on safety.

M28 CTS Specification 3.5.1.2 has Applicability "... at rated power ..." ITS Specification 3.3.3 has Applicability in MODES 1, 2, and 3. Since the ITS has broader MODE Applicability, this change is more restrictive and has no adverse impact on safety.

M29 The CTS is revised to adopt the ITS Specification 3.3.3 ACTIONS "Note 1." This Note excludes the MODE change restriction of LCO 3.0.4. Since no similar Notes exist in the CTS, this change imposes new requirements and is therefore more restrictive and has no adverse impact on safety.

M30 CTS Table 3.5-5 requires that, with both containment high range radiation monitoring channels inoperable, one of the channels be restored to OPERABLE status within 7 days, or a special report be prepared and submitted to the NRC within the following 14 days, detailing the cause of the inoperable channels and the action being taken to restore a channel to OPERABLE status. ITS Specification 3.3.3 requires that, if the 7 day Completion Time is not met, the unit be in MODE 3 within 6 hours and MODE 4 within 12 hours. Since this change requires a unit shutdown when both channels are inoperable and at least one cannot be returned to OPERABLE status within 7 days, it is more restrictive and has no adverse impact on safety.

M31 CTS Table 3.5-5, Note 6, requires that with both containment hydrogen monitoring channels inoperable, that one channel be restored to OPERABLE status within 14 days. ITS LCO 3.3.3, Required Action E, requires that one channel be restored to OPERABLE status within 72 hours. This change imposes more restrictive requirements, and has no adverse impact on safety.

M32 The CTS is revised to adopt the following Functions from the plant specific Regulatory Guide 1.97 analysis in ITS Specification 3.3.3: Steam Generator (SG) Pressure and Level, Containment Spray Additive Tank Level, Containment Isolation Valve Position Indication, Power Range and Source Range Neutron Flux, Reactor Coolant System (RCS) Pressure, RCS Hot and Cold Leg Temperature, Refueling Water Storage Tank Level, and Condensate Storage Tank Level. Since no similar Specifications or requirements exist in the CTS, this change imposes new requirements and is therefore more restrictive and has no adverse impact on safety.

M33 CTS Table 3.5-5, Note 8 requires that at least one thermocouple be restored to OPERABLE status within a specified time, or be in Hot Shutdown within the next 12 hours and < 350°F within the next 30 hours. ITS Specification 3.3.3 Required Action G requires that, under those

circumstances, the unit be placed in MODE 3 in 6 hours, and in MODE 4 in 12 hours. This change imposes shorter Completion Times, and is therefore more restrictive and has no adverse impact on safety.

M34 CTS Table 4.1-1 is revised to adopt Surveillance Requirements SR 3.3.3.1 and SR 3.3.3.2 for the Power Operated Relief Valve (PORV), PORV block valve, and pressurizer safety valve position indicators. SR 3.3.3.1 requires performance of a monthly CHANNEL CHECK, and SR 3.3.3.2 requires performance of a CHANNEL CALIBRATION at a Frequency of 18 months. This change imposes new requirements, and is therefore more restrictive and has no adverse impact on safety.

M35 The CTS is revised to adopt ITS Specification 3.3.4, "Remote Shutdown System," in the ITS. Since no similar Specification exists, this change is more restrictive and has no adverse impact on safety.

M36 CTS Table 3.5-3, Functional Units 3.A (loss of voltage protection) and 3.B (degraded voltage protection), have Applicability in the condition, "Reactor Critical." ITS Specification 3.3.5 has Applicability in MODES 1, 2, 3, and 4; and when associated Diesel Generator (DG) is required to be OPERABLE by LCO 3.8.2, "AC Sources-Shutdown and During Movement of Irradiated Fuel Assemblies." This change imposes more restrictive requirements, and has no adverse impact on safety.

M37 The CTS is revised to adopt ITS Specification 3.3.5 Required Action C. Required Action C requires that, with one or more Functions with two or more channels per bus inoperable, all but one channel be restored to OPERABLE status in 1 hour. Adoption of this Required Action imposes more restrictive requirements, and has no adverse impact on safety.

M38 CTS Table 3.5-4, ACTION 15, requires that with certain instrumentation channels inoperable, power operation may continue provided the Containment Ventilation Purge and Exhaust valves are maintained closed. ITS Specification 3.3.6, Required Action A, requires containment purge supply and exhaust valves be closed immediately or to enter the applicable conditions and Required Actions of LCO 3.9.3, "Containment Penetrations." Adoption of Required Action A.2 imposes more restrictive requirements, and has no adverse impact on safety.

M39 Not Used.

M40 The CTS Table 3.5-4, Ventilation Isolation Function, has Applicability "during containment purge." ITS Specification 3.3.6 has Applicability During Purging; during CORE ALTERATIONS, and during movement of irradiated fuel assemblies within containment. Since this change imposes broader Applicability requirements, it is more restrictive and has no adverse impact on safety.

M41 The CTS is revised by adopting ITS Table 3.3.6-1 Functions 1 and 2, which specify operability requirements for Manual Initiation and Automatic Actuation Relays. Since this change imposes additional operability requirements, it is more restrictive and has no adverse impact on safety.

M42 The CTS is revised to adopt ITS Surveillance Requirements SR 3.3.6.1, SR 3.3.6.2, SR 3.3.6.3 (and Note), and SR 3.3.6.4, which provide requirements to assure OPERABILITY of the containment ventilation isolation Function. Since no similar Specifications exist, this change is more restrictive and has no adverse impact on safety.

M43 The CTS is revised to adopt ITS Specification 3.3.7, "CREFS Actuation Instrumentation." Since no similar Specification exists, this change is more restrictive and has no adverse impact on safety.

M44 CTS Table 3.4-1, Note 2, requires an inoperable channel to be restored to OPERABLE status within 48 hours, or commence a normal plant shutdown and cooldown to ≤ 350°F. ITS Specification 3.3.4, Condition D requires that the inoperable channel be restored to OPERABLE status in 48 hours, or be in MODE 3 in 54 hours. Condition D requires further that the unit be in MODE 4 in 60 hours. Since this change imposes Completion Time restrictions where none exist, it is more restrictive and has no adverse impact on safety.

M45 CTS Specification 3.10.5.2 permits one RTB bypass breaker to be racked in and closed for up to 12 hours. ITS Specification 3.3.1 ACTION Q "Note", and ACTION R "Note 1," permit one train to be bypassed for up to 12 hours for surveillance testing provided the other train is OPERABLE. A new ACTION R "Note 2" is adopted, which permits one RTB to be bypassed for up to 2 hours for maintenance on undervoltage or shunt trip mechanisms, provided the other train is OPERABLE. Since the CTS is silent on permissible reasons for racking in and closing the RTB bypass breaker, and the ITS restricts the 12 hour bypass time to surveillance testing, and provides only 2 hours for maintenance activities, this change is more restrictive and has no adverse impact on safety.

M46 The CTS is revised to adopt a Containment Radioactivity High setpoint for the R-11 and R-12 containment monitors to initiate containment isolation. Since no other similar specification exists, this change is more restrictive and has no adverse impact on safety.

M47 CTS Table 3.5-5, Item 11, is revised in ITS Table 3.3.3-1, Item 11 to specify two OPERABLE channels of containment hydrogen monitors. The overall effect of this change is to limit the allowed outage time for one inoperable channel to 30 days in accordance with ITS 3.3.3 Required Action A.1, at which time a report to the NRC is required. This change is more restrictive and has no adverse impact on safety.

M48 CTS Table 3.4-1, Note 1, requires that, if either 4 kV undervoltage relay fails, the equivalent of an undervoltage signal must be inserted in the steam driven AFW pump start circuit within 4 hours; the affected relay must be repaired within 7 days, or commence a normal plant shutdown to hot standby. ITS Specification 3.3.8, Required Action B, requires under similar conditions, that the inoperable channel be placed in trip in 6 hours, or be in MODE 3 in 12 hours, and MODE 4 in 18 hours. These are more restrictive requirements with respect to failure to meet the allowed outage time of 6 hours, and has no adverse impact on safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS Specification 2.3.1.2.d provides a descriptive definition of a function generated by the lead-lag controller for T_{avg} dynamic compensation, a discussion of the definition of f(ΔI), and how the permissible flux difference range is extended for variations in power level, all in the overtemperature ΔT calculation. CTS Specification 2.3.1.2.e provides a descriptive definition of a function generated by the rate-lag controller for T_{avg} dynamic compensation, and definition of a time constant in the rate-lag controller for T_{avg}, all in the overpower ΔT calculation. These details related to the overtemperature ΔT and overpower ΔT functions are not retained in the ITS and are relocated to licensee controlled documents.

The details associated with the involved Specifications are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for OPERABILITY of the overtemperature ΔT and overpower ΔT functions. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

LA2 CTS Specification 2.3.3 requires that the RCS narrow range temperature sensor response time be less than or equal to a 4.0 second lag time constant. This detail is not retained in the ITS and is relocated to licensee controlled documents.

The details associated with the involved Specification are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for OPERABILITY of the RCS temperature sensors. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

LA3 CTS Specification 3.10.5.1.b, requires that the reactor not be made critical unless the reactor trip bypass breakers are racked out. This detail is not retained in the ITS and is relocated to licensee controlled documents.

The details associated with the involved Specification are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for OPERABILITY of the Reactor Trip System. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

LA4 CTS Table 4.1-1 contains instrument channel surveillance requirements for Analog Rod Position, Rod Position Bank Counters, Charging Flow, Residual Heat Removal (RHR) Pump Flow, Boric Acid Tank Level, Refueling Water Storage Tank (RWST) Level, Volume Control Tank Level, Containment Pressure, Boric Acid Makeup Flow Channel, Accumulator Level and Pressure, Steam Generator Pressure, and RCS High Point Vents. This detail is not retained in the ITS and is relocated to licensee controlled documents.

The details associated with the involved Specifications are not required to be in the ITS to provide adequate protection of the public health and safety, because these details are instrument channel surveillance requirements that are not associated with any Limiting Condition for Operation. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

LA5 CTS Table 3.5-1, Item 7, requires that the Containment Radioactivity High alarm be set with a method described in the ODCM. This detail is not retained in the ITS and is relocated to licensee controlled documents.

The details associated with the involved Specifications are not required to be in the ITS to provide adequate protection of the public health and safety, because the ITS still retains the requirements for OPERABILITY of the R-11 and R-12 containment monitors. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

LA6 CTS Table 3.5-5, Note 5, requires a pre-planned alternate method of monitoring be available before the Required Action to restore both inoperable post accident monitoring channels to OPERABLE status within 7 days is allowed. This detail is not retained in the ITS and is relocated to licensee controlled documents.

The details associated with the involved Specifications are not required to be in the ITS to provide adequate protection of the public health and safety, because the ITS still retains the allowed outage times and requirements for OPERABILITY of the post accident monitoring channels. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

LA7 CTS Table 3.5-2, Item 15 and associated Actions 9 and 10 provide requirements for the Control Rod Misalignment Monitor consisting of an ERFIS Rod Position Deviation Monitor and a Quadrant Power Tilt Monitor. The Control Rod Misalignment Monitor provides an alarm should an individual rod position deviate from its group's position or if quadrant power tilt ratio exceed limits. The Control Rod Misalignment Monitor does not provide input into the Reactor Trip System or the Rod Control System. Should an ERFIS Rod Position Deviation Monitor become inoperable with the reactor critical, additional periodic operator logging requirements are specified after exceeding specified magnitudes of either changes in load or rod motion. Should an Quadrant Power Tilt Monitor become inoperable with the reactor > 50 RTP, additional periodic operator logging requirements are specified after exceeding specified magnitudes of either changes in load or rod motion. Additionally, should both the Rod Position Deviation Monitor and Quadrant Power Tilt Monitor become inoperable for 2 hours or longer with reactor power > 50 RTP, the nuclear overpower trip setpoint is required to be reset to ≤ 93% RTP. The requirements associated with this specification are not retained in the ITS and are relocated to licensee controlled documents.

The requirements associated with the involved Specifications are not required to be in the ITS to provide adequate protection of the public health and safety, because the ITS still retains appropriate requirements for both Rod Group Alignment Limits and Quadrant Power Tilt Ratio. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this requirement is acceptable.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 The CTS is revised to adopt the "ALLOWABLE VALUE" column from ISTS Table 3.3.1-1 and Table 3.3.2-1. This column is added to provide an allowance for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RPS and ESFAS channels that must function in harsh environments. The Allowable Values specified in Table 3.3.1-1 and Table 3.3.2-1 are conservatively set with respect to the analytical limits. In establishing these allowable values, some have been determined to be less conservative than the CTS trip setpoint limits. The less conservative parameters, which include power range neutron flux (high and low), OTΔT, OPΔT, low pressurizer pressure, and RCS loop low flow, are considered to be a relaxation of requirements, which is less restrictive.

This change is acceptable, however, because the actual nominal trip setpoint is more conservative than that specified by the Allowable Value to account for changes in random measurement errors, such as drift during a surveillance interval. Setpoints in accordance with the Allowable Value ensure that safety limits are not violated during abnormal operational occurrences (AOOs), and that the consequences of design basis accidents (DBAs) will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed. The Allowable Values listed in Table 3.3.1-1 and Table 3.3.2-1 are conservatively set with respect to the analytical limits, and are based on the methodology described in the company setpoint methodology procedure. The magnitudes of uncertainties are factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes. This change is consistent with NUREG-1431.

L2 CTS Specifications 2.3.1.2.d and 2.3.1.2.e set the values of certain OTΔT and OPΔT parameters as being "=" to specific values. ITS Table 3.3.1-1, Notes 1 and 2, set these same parameters as being either "≥" or "≤" specific values. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because these parameter settings are cycle specific and only permit installation of a more restrictive setpoint in the actual hardware. Although these parameters normally do not change, they are subject to modification as a result of a reload safety analysis. This change is consistent with NUREG-1431.

L3 CTS Table 3.5-2 Table Notation ACTION 6 permits operation to proceed until performance of the next required operational test, provided that the inoperable channel be placed in the tripped condition within 1 hour. ITS Specification 3.3.1 ACTION E permits instead, unrestricted continued operation provided that the inoperable channel be placed in trip in 6

hours. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because the 6 hours allowed to place the inoperable channel in trip is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990. Additionally, placing the inoperable channel in trip results in a partial trip condition requiring only one-out-of-two logic for actuation. This change is consistent with NUREG-1431.

L4 CTS Table 3.5-2 inoperable channel ACTION for 4kV Undervoltage requires placing the inoperable channel in trip in 1 hour, restoring the channel to OPERABLE status in 7 days or placing the unit in the hot shutdown condition in the next 8 hours. ITS Table 3.3.1-1 ACTION M for the same function requires placing the inoperable channel in trip in 6 hours, or reducing THERMAL POWER to less than P-7 in 12 hours. This change can be considered a relaxation of requirements due to the additional time permitted to place the inoperable channel in trip and elimination of the specific shutdown requirement, and is less restrictive. This change is acceptable, however, because the 6 hour time to place the inoperable channel in trip is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990, and considers the redundant capability provided by the remaining OPERABLE channel, and the unlikelihood of occurrence of an event that may require the protection afforded by the function during this period. This change is consistent with NUREG-1431.

L5 CTS Table 3.5-2 Table Notations "****" and "*****" relate to MODE Applicability when certain reactor trip functions are above either the P-10 or P-7 setpoints. ITS Table 3.3.1-1 Footnotes (f) and (h) relate to MODE Applicability when certain reactor trip functions are above only the P-7 setpoint. Elimination of the "or" connector (with P-10) is a relaxation of requirements, which is less restrictive, since the P-10 setpoint can be exceeded without the trip function having MODE Applicability. This change is acceptable, however, since both the P-7 and P-10 setpoints are at 10 percent RTP, and with a reactor trip function above the P-7 setpoint, the unit is in essentially the same condition. This change is consistent with NUREG-1431.

L6 CTS Table 3.5-2 ACTION 1 requires an inoperable manual reactor trip function to be restored to OPERABLE status within 12 hours, or be in hot shutdown within the next 8 hours. ITS Specification 3.3.1 ACTION B requires an inoperable manual reactor trip function to be restored to OPERABLE status within 48 hours, or be in MODE 3 within 54 hours, and the RTBs open in 55 hours. While the adoption of the requirement to open the RTBs in 55 hours is a new, more restrictive requirement, the overall change in Completion Times is a relaxation of requirements, and is less restrictive. The 48 hour Completion Time is acceptable, however, considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval. The 6 additional hours to reach MODE 3 from full power in an orderly manner and without

challenging unit systems are reasonable, based on operating experience. This change is consistent with NUREG-1431.

L7 The CTS is revised to adopt ISTS Specification 3.3.1, Required Action D.2.2 "Note." CTS Table 3.5-2, ACTION 2, requires under certain conditions, that the QPTR be monitored every 12 hours. The D.2.2 Note only requires this Surveillance to be performed when the Power Range Neutron Flux input to QPTR is inoperable. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because failure of a component in the Power Range Neutron Flux channel which renders the High Flux Trip Function inoperable may not affect the capability to monitor QPTR. As such, performing the Surveillance using the movable incores is redundant, and not necessary. This change is consistent with NUREG-1431.

L8 CTS Table 3.5-2 ACTION 2 requires an inoperable channel be placed in trip within 1 hour, and <u>either</u>: a) power reduced to ≤ 75% RTP <u>and</u> power range flux trip setpoint reduced to ≤ 85% RTP in 4 hours <u>or</u>: b) QPTR be monitored every 12 hours. ITS Specification 3.3.1 ACTION D requires <u>either</u>: a) the inoperable channel be placed in trip within 6 hours <u>and</u> power reduced to ≤ 75% RTP in 12 hours, <u>or</u> b) the inoperable channel be placed in trip within 6 hours <u>and</u> SR 3.2.4.2 (QPTR) be performed once per 12 hours, <u>or</u> c) be in MODE 3 in 12 hours. The differences here are discussed from the perspective of the most and least restrictive actions that can be taken in response to the CONDITION of an inoperable power range neutron flux - high channel. The least restrictive actions that can be taken in the CTS are to place the channel in trip in 1 hour and monitor QPTR at a Frequency of 12 hours. The least restrictive actions that can be taken in the ITS are to place the channel in trip in 6 hours and monitor QPTR (SR 3.2.4.2) at a Frequency of 12 hours. The ITS Frequency of 6 hours for placing the channel in trip is a relaxation of requirements, and is a less restrictive change. Elimination of the requirement to reduce the high neutron flux trip setpoint to ≤ 85% is also a less restrictive change. This change is acceptable, however, because the 6 hour Frequency for placing the channel in trip is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990, and not reducing the high neutron flux trip setpoint to ≤ 85% has no adverse impact on the remaining OPERABLE power range neutron flux channels maintaining their capability to prevent the core from operating in an overpower condition. While reduction of the trip setpoint would limit the overshoot in a power excursion, maintaining the high flux trip at its normal setpoint still provides adequate protection in the event of a power excursion. This change is consistent with NUREG-1431.

L9 The CTS is revised to adopt ISTS Specification 3.3.1 Required Action G in the ITS. The CTS has no specific action requirements in the event two Intermediate Range Neutron Flux channels become inoperable when the unit is operating at a THERMAL POWER >P-6 and <P-10. CTS Section 3.0 would therefore be entered, requiring the unit to be in hot shutdown in

8 hours, and in cold shutdown within the next 30 hours. ISTS Required Action G requires, under these conditions, that operations involving positive reactivity additions be suspended immediately, and THERMAL POWER be reduced to <P-6 in 2 hours. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because above the P-6 setpoint and below the P-10 setpoint, the intermediate range performs a monitoring function. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase. Power must also be reduced below the P-6 setpoint. Below P-6, the Source Range Neutron Flux channels will be able to monitor core power. This change is consistent with NUREG-1431.

L10 CTS Table 3.5-2 inoperable channel ACTION for low pressurizer pressure, high pressurizer water level, low reactor coolant flow (single loop and two loops), and 4kV underfrequency provides that operation may proceed until performance of the next required operational test if the inoperable channel is placed in trip in 1 hour. ITS Specification 3.3.1, Required Actions M and N for the same functions, require the inoperable channel be placed in trip in 6 hours or THERMAL POWER is reduced to less than P-7 and P-8, respectively, in 12 hours. This change can be considered a relaxation of requirements due to the additional time permitted to place the inoperable channel in trip and the addition of the option to reduce power to below P-7 as a Required Action, and is less restrictive. This change is acceptable, however, because the 6 hour time to place the inoperable channel in trip is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990, and considers the redundant capability provided by the remaining OPERABLE channel, and the unlikelihood of occurrence of an event that may require the protection afforded by the function during this period. Also, reduction of THERMAL POWER below the P-7 setpoint assures that the Function is out of the Applicability for which the Function is required. This change is consistent with NUREG-1431.

L11 CTS Table 3.5-2 inoperable channel ACTION for turbine trip on auto stop oil pressure and turbine stop valve closure provides that operation may proceed until performance of the next required operational test if the inoperable channel is placed in trip in 1 hour. ITS Specification 3.3.1 inoperable channel ACTION P for the same functions require the inoperable channel be placed in trip in 6 hours or THERMAL POWER is reduced to less than P-7 in 10 hours. This change can be considered a relaxation of requirements due to the additional time permitted to place the inoperable channel in trip, and is less restrictive. This change is acceptable, however, because the 6 hour time to place the inoperable channel in trip is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990, and considers the redundant capability provided by the remaining redundant OPERABLE channel, and the unlikelihood of occurrence of an event that may require the protection afforded by the function during this period. This change is consistent with NUREG-1431.

L12 CTS Table 3.5-2 Items 13 (4kV Underfrequency) and 14 (4kV Undervoltage) have Applicability Conditions of "Reactor Critical." ITS Table 3.3.1-1 Items 11 (Undervoltage RCPs) and 12 (Underfrequency RCPs) have Applicability in MODE 1, above the P-7 interlock. Since this change narrows the MODE of Applicability, it is a relaxation of requirements and is less restrictive. This change is acceptable, however, because below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked, since no conceivable power distributions could occur that would cause a DNB concern at that low power level. This change is consistent with NUREG-1431.

L13 CTS Table 3.5-3 is revised to add ITS Table 3.3.2-1 Items 1.b, 2.b, 3.a(2), 3.b(2), 4.b and 5.a, and add ACTIONS C and G. These items relate to the Automatic Actuation Logic and Actuation Relays for various ESFAS Functions. The CTS does not explicitly identify the requirement for OPERABILITY of Automatic Actuation Logic and Actuation Relays. Consequently, inoperability of the Automatic Actuation Logic and Actuation Relays results in entry into CTS Section 3.0 with the requirement to achieve hot shutdown in 8 hours and cold shutdown within an additional 30 hours. The identification of the Automatic Actuation Logic and Actuation Relays within the LCO, and the addition of Required Actions C and G result in a relaxation of requirements by providing an allowed outage time of 6 hours for the Automatic Actuation Logic and Actuation Relays before a shutdown is required. The overall effect of this change is therefore less restrictive. This change is acceptable, however, because the redundant train provides trip capability during the allowed outage time. Also, the 6 hour Completion Time is further justified based on the unlikelihood of an event occurring during this interval. This change is consistent with NUREG-1431.

L14 CTS Table 4.1-1, Item 1 (Nuclear Power Range) and Item 4 (Reactor Coolant Temperature) require channel functional tests be performed on a bi-weekly frequency. ITS SR 3.3.1.7 and SR 3.3.1.8 require performance of a COT at a Frequency of 92 days. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because the change from a 31 day to 92 day Frequency is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990, and the 14 day CTS Frequency is adequately bounded by the analysis of the 31 day Frequency. This change is consistent with NUREG-1431.

L15 CTS Table 4.1-1, Item 2 (Nuclear Intermediate Range) and Item 3 (Nuclear Source Range), require functional testing prior to each reactor startup if a functional test has not been performed in the previous 7 days. ITS SR 3.3.1.8 requires that a COT be performed prior to reactor startup if the COT has not been performed in the previous 92 days. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because the change from a 7 day to 92 day Frequency

is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990, and the 7 day CTS Frequency is adequately bounded by the analysis of the 31 day Frequency. This change is consistent with NUREG-1431.

L16 CTS Table 4.1-1, Items 5 (Reactor Coolant Flow), 6 (Pressurizer Water Level), 7 (Pressurizer Pressure), 8 (4kV Voltage), 11 (Steam Generator Level), 39 (Steam/Feedwater Flow Mismatch), and 40 (Low Steam Generator Water Level) require monthly testing. ITS SR 3.3.1.7 and SR 3.3.1.9 are performed on a Frequency of 92 days. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because the change from a 31 day to 92 day Frequency is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990. This change is consistent with NUREG-1431.

L17 CTS Table 4.1-1, Item 25, Turbine First Stage Pressure, requires a functional test on a monthly Frequency. ITS SR 3.3.1.13 requires performance of a COT on an 18 month Frequency. This is a relaxation of requirements and is less restrictive. This change is acceptable, however, since the turbine first stage pressure signal is used only during unit startup to feed the P-7 permissive interlock, which bypasses

the high pressurizer level, low pressurizer pressure, RCS low flow, and turbine trip reactor trips below 10% RTP, which is an infrequent operation. This change is consistent with NUREG-1431.

L18 CTS Table 4.1-1, Item 27, Logic Channel Testing, requires monthly functional testing during hot shutdown and power operations, and for the source range channels prior to each reactor startup, if not performed within the previous 7 days. ITS Surveillance Requirements SR 3.3.1.5 and SR 3.3.2.2 require an ACTUATION LOGIC TEST be performed at a Frequency of 31 days on a STAGGERED TEST BASIS. Since each channel will only be tested every 62 days, this is a relaxation of requirements and is less restrictive. This change is acceptable, however, because the Frequency of 31 days on a STAGGERED TEST BASIS is based on industry operating experience, considering instrument reliability and operating history data. This change is consistent with NUREG-1431.

L19 CTS Table 4.1-1, Item 30, Reactor Trip Breakers, requires that a functional test be performed on a monthly frequency. ITS SR 3.3.1.4 requires that a TADOT be performed at a Frequency of 31 days on a STAGGERED TEST BASIS. Since each RTB will now be tested every 62 days, this is a relaxation of requirements and is less restrictive. This change is acceptable, however, because the Frequency of 31 days on a STAGGERED TEST BASIS is based on industry operating experience, considering instrument reliability and operating history data. This change is consistent with NUREG-1431.

L20 CTS Table 4.1-1, Item 47, Reactor Trip Bypass Breakers, requires that a functional test be performed at monthly (M) frequency. ITS SR 3.3.1.4

requires that a TADOT be performed prior to placing the bypass breaker in service. Since the bypass breakers are only placed in service when the RTBs are being tested, and each RTB will now be tested every 62 days, this is a relaxation of requirements and is less restrictive. This change is acceptable, however, because the Frequency of 31 days on a STAGGERED TEST BASIS is based on industry operating experience, considering instrument reliability and operating history data. This change is consistent with NUREG-1431.

L21 CTS Table 3.5-3 ACTION 12 permits power operation to continue until performance of the next required operational test, provided the inoperable channel is placed in trip within 1 hour. ITS Specification 3.3.2, Conditions D and E permit power operation to continue, with limits, provided the inoperable channel is placed in trip in 6 hours. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because Conditions C and D generally apply to functions that operate on two-out-of-three logic, and failure of one channel would place the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration, which satisfies redundancy requirements. Also, the 6 hour Completion Time is further justified based on the unlikelihood of an event occurring during this interval. This change is consistent with NUREG-1431.

L22 CTS Table 3.5-5 is revised by adopting ISTS Specification 3.3.3 Conditions A, B, C, and H. This change will require that, with one containment high range radiation monitoring channel inoperable, the channel be restored to OPERABLE status within 30 days, instead of 7. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, considering the low probability of an event requiring use of this Function during this time interval, and there is an installed OPERABLE redundant channel. This change is consistent with NUREG-1431.

L23 CTS Table 3.5-5, Note 1 requires that, if one AFW flow indicator becomes inoperable, it must be restored to OPERABLE status within 7 days. ITS Specification 3.3.3, Required Action A requires an inoperable AFW flow indicator to be restored to OPERABLE status within 30 days. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because the 30 day Completion Time takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from the instrument), and the low probability of an event requiring PAM instrumentation during this interval.

L24 CTS Table 3.5-5, Note 6, requires that with both containment hydrogen monitoring channels inoperable, and one channel cannot be restored to

OPERABLE status within the specified time, the unit be placed in Hot Shutdown within 6 hours and ≤200°F within the following 30 hours. ITS Specification 3.3.3, Required Action E requires under the same conditions, that the unit be placed in MODE 3 within 6 hours, and in MODE 4 within the following 6 hours. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, considering the backup capability of the Post Accident Sampling System to monitor hydrogen concentration for evaluation of core damage and to provide information for operator decisions; and the unlikelihood of an event that would require use of the hydrogen monitors in the interval. This change is consistent with NUREG-1431.

L25 CTS Table 3.5-5, Note 8 requires that the inoperable thermocouples be restored to OPERABLE status within 7 days, or be in Hot Shutdown within the next 12 hours and < 350°F within the next 30 hours. ITS Specification 3.3.3, Required Actions A and B require that the inoperable channel be restored to OPERABLE status within 30 days, or the reporting requirements of ITS Specification 5.6.6 be initiated immediately. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, considering the unlikelihood of an event occurring in the extended interval, together with a redundant OPERABLE channel available within the same quadrant. This change is consistent with NUREG-1431.

L26 CTS Table 3.5-5, Note 8 requires that at least one thermocouple be restored to OPERABLE status within 48 hours. ITS Specification 3.3.3, Required Action C requires that one inoperable channel be restored to OPERABLE status within 7 days. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, considering the unlikelihood of an event occurring in the extended interval. This change is consistent with NUREG-1431.

L27 CTS Table 4.1-1 requires that the Containment Level, Pressure, Hydrogen and Radiation Monitors be tested at an "R" Frequency. ITS Specification 3.3.3 has no such requirement. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because a CHANNEL CALIBRATION is performed on these channels at an 18 month Frequency. The CHANNEL CALIBRATION encompasses all the testing requirements for these Functions, from sensor to indicator. This change is consistent with NUREG-1431.

L28 CTS Table 3.5-3, Functional Unit 3.A, Action 14 requires that, with the number of OPERABLE Loss of Voltage channels one less than the Total Number of channels, the inoperable channel be placed in block within 1 hour and be restored to OPERABLE status within 48 hours, or the reactor be placed in hot shutdown within the next 8 hours and cold shutdown within the following 30 hours.

ITS Specification 3.3.5 requires that, if one or more Loss of Voltage Functions have one channel per bus inoperable, the inoperable channel be placed in block in 6 hours. If that Completion Time is not met, the applicable Condition(s) and Required Action(s) for the associated DG made inoperable by the loss of power instrumentation be entered immediately. This is a relaxation of requirements, and is less restrictive.

This change is acceptable, however, because there are two Loss of Voltage channels per bus, which are configured in a one-out-of-one logic, such that if either channel sees loss of voltage, it will trip the bus. With one channel placed in a blocked condition, the OPERABLE channel is still available to trip the bus. The 6 hour Completion Time is acceptable, considering the Function remains OPERABLE on both emergency busses and the low probability of an event occurring during this interval. This change is consistent with NUREG-1431.

L29 CTS Table 3.5-3, Functional Unit 3.B, Action 14 requires that, with the number of OPERABLE Degraded Voltage channels one less than the Total Number of channels, the inoperable channel be placed in block within 1 hour and be restored to OPERABLE status within 48 hours, or the reactor be placed in hot shutdown within the next 8 hours and cold shutdown within the following 30 hours.

ITS Specification 3.3.5 requires that, if one or more Degraded Voltage Functions have one channel per bus inoperable, the inoperable channel be placed in trip in 6 hours. If that Completion Time is not met, the applicable Condition(s) and Required Action(s) for the associated DG made inoperable by the loss of power instrumentation be entered immediately. This is a relaxation of requirements, and is less restrictive.

This change is acceptable, however, because there are three Degraded Voltage channels per bus, which are configured in a two-out-of-three logic, such that if any two channels see a degraded voltage condition, they will trip the bus. With one channel placed in a tripped condition, the two OPERABLE channels are still available to trip the bus in a one-out-of-two logic arrangement. The 6 hour Completion Time is acceptable, considering the Function remains OPERABLE on both emergency busses and the low probability of an event occurring during this interval. This change is consistent with NUREG-1431.

L30 CTS Specification 3.5.1 is revised to adopt the ISTS Specification 3.3.5 "Note" to Required Action B.1 in the ITS. The Note permits an inoperable Degraded Voltage Function channel to be bypassed for up to 4 hours for surveillance testing of other channels. Adoption of this Note constitutes a relaxation of requirements, and is therefore less restrictive. This change is acceptable, however, because there are three Degraded Voltage channels per bus, and this allowance is made

where bypassing the channel does not cause an actuation, and where at least two other channels per bus are monitoring the parameter. The Degraded Voltage Function is arranged in a two-out-of-three configuration. Bypassing one channel would still provide a two-out-of-two logic. The time allowed is reasonable, considering the Function remains fully OPERABLE on each bus and the low probability of an event occurring during the interval. This change is consistent with NUREG-1431.

L31 CTS Specification 3.8.1.b requires that the radiation monitors which initiate containment ventilation isolation be tested and verified to be OPERABLE immediately prior to refueling operations. This requirement is not retained in the ITS. This constitutes a relaxation of requirements, and is therefore less restrictive. This change is acceptable, however, because the radiation monitors are demonstrated OPERABLE at a Frequency of 92 days by performance of a CHANNEL OPERATIONAL TEST. This change is consistent with NUREG-1431.

L32 CTS Table 3.4-1, Function 1, requires under certain channel inoperability conditions, that the unit be maintained in hot shutdown. ITS Specification 3.3.8, Required Action B, requires under similar conditions, that the inoperable channel be placed in trip in 6 hours, or be in MODE 3 in 12 hours, and MODE 4 in 18 hours. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because because placing the inoperable channel in trip maintains the AFW pump autostart Function OPERABLE, but in a one-out-of-two configuration, instead of two-out-of-three. The allowance of 6 hours to return the channel to OPERABLE status or place it in trip is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990. This change is consistent with NUREG-1431.

L33 CTS Table 3.4-1, Note 1, requires that, if either 4 kV undervoltage relay fails, the equivalent of an undervoltage signal must be inserted in the steam driven AFW pump start circuit within 4 hours; the affected relay must be repaired within 7 days, or commence a normal plant shutdown to hot standby. ITS Specification 3.3.8, Required Action B, requires under similar conditions, that the inoperable channel be placed in trip in 6 hours, or be in MODE 3 in 12 hours, and MODE 4 in 18 hours. This is a relaxation of requirements with respect to the 6 hour allowed outage time, and is less restrictive. This change is acceptable, however, because placing the inoperable channel in trip maintains the steam driven AFW pump autostart Function OPERABLE, but in a one-out-of-one configuration, instead of two-out-of-two. The allowance of 6 hours to return the channel to OPERABLE status or place it in trip is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990. This change is consistent with NUREG-1431.

L34 CTS Specification 3.5.1.4 requires the containment ventilation isolation function only when containment integrity is required. ITS Specification

3.3.6 requires that instrumentation for each function in Table 3.3.6-1 be OPERABLE during CORE ALTERATIONS and during movement of irradiated fuel within containment. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because the containment isolation function remains OPERABLE in MODES 1, 2, 3, and 4, upon initiation of a SI signal. Actuation on a high radiation signal from the R-11 or R-12 containment monitors is only required during CORE ALTERATIONS and during movement of irradiated fuel within containment, because no credit is taken for these instruments in the accident analyses, other than the fuel handling accident.

L35 CTS Table Note (*) for Table 3.5.2 are revised to add ". . . and rods not fully inserted or Rod Control System Capable of rod withdrawal," to and is incorporated into ITS Table 3.3.1-1, Note (a). This change reduces of Applicability in MODEs 3, 4, and 5 for Functions 1, 4, 18, 19, and 20 of ITS Table 3.3.1-1. This change relaxes requirements and is less restrictive. This change is acceptable because the remaining Applicability for Functions 1, 4, 18, 19, and 20 ensures that the reactor trip functions will be available when required. Specifically, the current licensing basis allows the rods to be five steps from the bottom with the lift disconnect switches open to prevent uncontrolled rod withdrawal. In this condition credit for the control and shutdown rods can be taken in the shutdown margin without relying on a reactor trip.

L36 CTS Table 3.5-2 ACTION 3 requires for an inoperable intermediate range neutron flux channel with THERMAL POWER above the P-6 setpoint, but below 10% RTP, that the inoperable channel be restored to OPERABLE status prior to increasing THERMAL POWER above 10% RTP. ITS Specification 3.3.1 ACTION F requires for an inoperable intermediate range neutron flux channel with THERMAL POWER above the P-6 setpoint, but below the P-10 setpoint, that THERMAL POWER either be reduced to below P-6 or increased above P-10 in 2 hours. The intermediate range neutron flux channels must be OPERABLE when the power level is above the capability of the source range and below the capability of the power range. The CTS does not permit an increase in power level to exit the Applicability of the intermediate range detectors. The Required Action to increase THERMAL POWER to exit the Applicability for the intermediate range detectors is less restrictive. The change is acceptable since the intermediate range detectors are not required to be OPERABLE above P-10 setpoint, and power range instrumentation provides the necessary protection above P-10. This change is consistent with NUREG-1431.

L37 The CTS is revised to adopt ISTS Specification 3.3.1 Required Action O in the ITS. The CTS has no specific action requirements in the event one Reactor Coolant Pump (RCP) breaker position channel is inoperable. CTS Section 3.0 would therefore be entered, requiring the unit to be in hot shutdown in 8 hours, and in cold shutdown within the next 30 hours. ISTS Required Action O requires, under these conditions, that the

channel be restored to OPERABLE status within 6 hours, or reduce THERMAL POWER to < P-8 in 10 hours. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because the allowed outage time of 6 hours granted by Required Action O is consistent with WCAP-P-A, Supplement 2, Rev. 1, June 1990; below the P-8 setpoint the RCP breaker position is not required to anticipate the RCS low flow trip to protect against DNB; the probability of an event requiring the Function of RCP breaker position is low during the allowed outage time; and, the most likely event for which the Function would be required is a loss of offsite power which would result in the trip of the remaining two RCPs, giving a signal to the RPS. This change is consistent with NUREG-1431.

RELOCATED SPECIFICATIONS

R1	Table 3.5-5	Item 3	RCS Subcooling Monitor
		Item 7a	Noble Gas Effluent Monitor - Main Steam Line
		Item 7b	Noble Gas Effluent Monitor - Main Vent Stack - High Range
		Item 7b	Noble Gas Effluent Monitor - Main Vent Stack - Mid Range
		Item 7c	Noble Gas Effluent Monitor - Spent Fuel Pit Lower Level - High Range
		Item 12	Reactor Vessel Level Instrumentation System (RVLIS)
		Note 2	
		Note 4	
		Note 7	
	Table 4.1-1	Item 34	RCS Subcooling Monitor
		Item 38a	Noble Gas Effluent Monitor - Main Steam Line
		Item 38b	Noble Gas Effluent Monitor - Main Vent Stack - High Range
		Item 38b	Noble Gas Effluent Monitor - Main Vent Stack - Mid Range
		Item 38c	Noble Gas Effluent Monitor - Spent Fuel Pit Lower Level - High Range
		Item 48	Reactor Vessel Level Instrumentation System (RVLIS)

These Specifications, or Limiting Conditions for Operation (Chapter 3.0), are not retained in the ITS because they have been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provided in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of each of these Specifications is provided in the report, "Application of Selection Criteria to the H. B. Robinson Steam Electric Plant Unit No. 2 Technical Specifications."

These Limiting Conditions for Operation, and their associated Surveillance Requirements (Chapter 4.0), are relocated to licensee controlled documents. Relocation of the specific requirements for systems or variables contained in these Specifications to licensee documents will have no impact on the operability or maintenance of those systems or variables. The licensee will initially continue to meet the requirements contained in the relocated Specifications. The licensee is allowed to make changes to these requirements in accordance with the provisions of 10 CFR 50.59. Such changes can be made without prior NRC

approval, if the change does not involve an unreviewed safety question, as defined in 10 CFR 50.59. These controls are considered adequate for assuring that structures, systems, and components in the relocated Specifications are maintained operable, and variables are maintained within limits. This change is consistent with the NRC Final Policy Statement on Technical Specification Improvements.

IMPROVED STANDARD TECHNICAL SPECIFICATION (ISTS) CONVERSION

CHAPTER 3.3 - INSTRUMENTATION

PART 3

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC), AND BASIS FOR CATEGORICAL EXCLUSION FROM 10 CFR 51.22

ADMINISTRATIVE CHANGES
("A" Labeled Comments/Discussions)

In the conversion of the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2 Technical Specifications to the proposed plant specific Improved Technical Specifications certain wording preferences or conventions are being adopted which do not result in technical changes (either actual or interpretational). Editorial changes, clarification, reformatting, rewording and revised numbering are being adopted to make the improved Technical Specifications consistent with NUREG-1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Administrative" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes consist of editorial changes and clarification, reformatting, rewording and renumbering of the current Technical Specifications. This process does not involve any technical changes to existing requirements. As such, these changes are administrative in nature and do not impact initiators of analyzed events or alter any assumptions relative to mitigation of accident or transient events. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. The proposed changes do not impose or eliminate any requirements. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed changes will not reduce a margin of safety because they do not impact any safety analysis assumptions. These changes are administrative in nature and, as such, do not impact any technical requirements. Therefore, these changes do not involve any reduction in a margin of safety.

MORE RESTRICTIVE CHANGES
("M" Labeled Comments/Discussions)

The HBRSEP Unit No. 2 Technical Specifications are proposed to be modified in some areas to impose more restrictive requirements than currently exist. These more restrictive changes are being imposed to be consistent with NUREG-1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "More Restrictive" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes provide requirements determined to be more restrictive than the current Technical Specifications requirements for operation of the facility. These more restrictive requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. These changes have been confirmed to ensure that no previously evaluated accident has been adversely affected. The more restrictive requirements being proposed enhance assurance that process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis of the unit. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. These changes do impose new or additional requirements which are consistent with assumptions made in the safety analysis and licensing basis. The additional requirements include new Surveillance Requirements, more restrictive Frequencies and Completion Times, new LCOs, more restrictive Required Actions and Applicabilities, and other operational restrictions that enhance safe operation. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact or increases the margin of plant safety. Each of the changes in this category, while providing new or additional requirements designed to

enhance plant safety, is consistent with the safety analyses and licensing basis. Therefore, these changes do not involve a reduction in a margin of safety.

LESS RESTRICTIVE-GENERIC CHANGES
("LA" Labeled Comments/Discussions)

In the conversion of the HBRSEP Unit No. 2 Technical Specifications to the proposed plant specific Improved Technical Specifications, portions of some Specifications which are descriptive in nature regarding equipment, systems, actions, surveillances or programs are proposed to be relocated from the Specifications to the Bases, Updated Final Safety Analysis Report, procedures or other licensee controlled documents. The details associated with the involved specifications are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for compliance with the applicable specifications. Changes to the Bases are controlled in accordance with the proposed Bases Control Program described in Chapter 5 of the Improved Technical Specifications. Changes to the UFSAR and administrative procedures which control revisions to these relocated requirements are controlled in accordance with licensee controlled programs.

This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Less Restrictive-Generic" and has concluded that they do not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that proposed changes do not involve a significant hazards consideration are discussed below.

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

   The proposed changes relocate requirements from the Technical Specifications to the Bases, Updated Final Safety Analysis Report, procedures or other licensee controlled documents. The documents containing the relocated requirements are subject to the change control of licensee controlled programs. Since any changes to these documents will be evaluated in accordance with the requirements of licensee controlled programs, no increase in the probability or consequences of an accident previously evaluated will be permitted without further NRC review. Therefore, these changes do not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes do not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. These changes do not introduce a new mode of

plant operation. Since any future changes to these requirements will be evaluated in accordance with licensee controlled programs, the possibility of a new or different kind of accident from any accident previously evaluated will not be permitted without further NRC review. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes will not reduce a margin of safety because they do not impact any safety analysis assumptions. The requirements that are transposed from the Technical Specifications to other licensee controlled documents are the same as the existing Technical Specifications. Since any future changes to these requirements will be evaluated in accordance with the requirements of licensee controlled programs, no reduction in any margin of safety will be permitted without further NRC review. Therefore, these changes do not involve any reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L1" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The Allowable Value is a limiting value that a trip setpoint may have, beyond which action must be taken such that the analytical value assumed in the accident analyses is not violated. The actual nominal trip setpoint is more conservative than that specified by the Allowable Value to account for changes in random measurement errors, such as drift, during a surveillance interval. Setpoints in accordance with the Allowable Value ensure that analytical limits are not violated during anticipated operational occurrences (AOOs), and that the consequences of design basis accidents (DBAs) will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The analytical limits of variables established by the safety analysis have not been changed. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The Allowable Values are based on a specific setpoint methodology which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint and Allowable Value. Sensors and signal processing equipment are assumed to operate within the allowances of these uncertainty magnitudes, thereby maintaining the margin to the safety limits. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L2" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The parameters involved include reference temperature and pressure settings, and certain time constants and other constants which are used in the continuous overtemperature ΔT and overpower ΔT calculations. These values normally do not change, but can be cycle specific, based on reload safety analyses. None of these parameters are considered initiators of accidents, since they are reference values based on plant design, and not actual values. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The overtemperature ΔT and overpower ΔT reference parameters and constants are not assumed to be initiators of accidents. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The parameters involved are reference parameters, which would only change based on a cycle reload analysis, power rerating, or other major design change, which would be incorporated in the accident analysis. Therefore, this change, which is consistent with the current analyses, does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L3" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change increases the time permitted to place an inoperable reactor trip Function channel in trip, and allows unlimited operation in that condition. Placing the channel in trip results in a partial trip condition, requiring only one-out-of-two logic for actuation, and the increased permitted time to place the inoperable channel in trip is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. This change increases the time permitted to place an inoperable reactor trip Function channel in trip, and allows unlimited operation in that condition. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   This change only extends the allowed time to place an inoperable reactor trip Function channel in trip, and allows unlimited operation in that condition. The extended time is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L4" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change increases the time permitted to place an inoperable 4kV undervoltage trip Function channel in trip, and eliminates a specific shutdown requirement, should the inoperable channel not be restored to OPERABLE status within the specified

time. The increased time to place the inoperable channel in trip considers the redundant capability provided by the remaining OPERABLE channel, and is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   This change extends the allowed time to place an inoperable channel in trip, and eliminates a specific shutdown requirement. The extended time is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1, and elimination of the specific shutdown requirement considers the redundant undervoltage trip capability provided by the remaining OPERABLE channel. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
<u>("L5" Labeled Comments/Discussions)</u>

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. Both the P-7 and P-10 permissive setpoints are actuated at approximately 10 percent RTP, and with the reactor trip Functions enabled above the P-7 setpoint, the unit is fully protected from high neutron flux condition. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal

plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   Since the unit is maintained fully protected from a high neutron flux condition, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L6" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The extended time interval to restore the inoperable manual reactor trip function to OPERABLE status considers that there are two automatic actuation trains and another manual actuation channel OPERABLE, and the low probability of an event occurring during this interval. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   Since the unit is maintained fully protected with two automatic actuation trains and another manual actuation channel OPERABLE, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L7" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The Surveillance Requirement "Note" identifies that failure of a component in a Power Range Neutron Flux channel which renders the High Flux Trip Function inoperable may not necessarily affect the capability to monitor QPTR, and therefore only requires performance of the QPTR SR, using the incore detectors, when Power Range Neutron Flux input to QPTR is inoperable. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L8" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The extended time interval permitted in the ITS to place the inoperable Power Range Neutron Flux channel in trip is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990. Not reducing the High Neutron Flux Trip Setpoint to ≤ 85% has no adverse impact, considering that there are three remaining OPERABLE channels, requiring only one-out-of-three logic for actuation. While reduction of the trip setpoint would limit the overshoot in a power excursion, maintaining the high flux trip at its normal setpoint still provides adequate protection in the event of a power excursion. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. The High Neutron Flux Trip remains OPERABLE, requiring a one-out-of-three logic to actuate. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L9" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The Intermediate Range Neutron

Flux channels are not assumed to be initiators of accidents. Suspension of all positive reactivity additions precludes any power level increase, and reducing power to below the P-6 setpoint puts the reactor in a condition where the Source Range Neutron Flux channels will monitor core power. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Inoperable instrument channels cannot initiate a new or different kind of accident. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L10" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The extended time interval permitted in the ITS to place the inoperable reactor trip Function channel in trip considers the redundant capability of the remaining OPERABLE channel, and is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Single channel trip capability is provided by the remaining redundant OPERABLE channel. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L11" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The extended time interval permitted in the ITS to place the inoperable reactor trip Function channel in trip considers the redundant capability of the remaining OPERABLE channel, and is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Single channel trip capability is provided by the remaining redundant OPERABLE channel. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L12" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The Undervoltage Reactor Coolant Pump (RCP) and Underfrequency RCP Functions do not have to be OPERABLE below the P-7 setpoint because there are no loss of flow trips below P-7. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L13" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The Automatic Actuation Logic and Actuation Relays Functions are required to be OPERABLE for Engineered Safety Features Actuation Systems (ESFAS) to be OPERABLE. This change specifically identifies the OPERABILITY requirement for ESFAS Automatic Actuation Logic and Actuation Relays and provides an allowed outage time of 6 hours. During the allowed outage time, the redundant train of Automatic Actuation Logic and Actuation Relays is available to perform the required function if required. The probability of an event requiring the ESFAS Function during this period is low. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change permits an allowed outage time for the Automatic Actuation Logic and Actuation Relays, and this change reduces the implied margin of safety associated with allowance of only a single train of Automatic Actuation Logic and Actuation Relays for 6 hours. The probability of an event requiring the Function during the allowed outage time is low, and the redundant train of Automatic Actuation Logic and Actuation Relays is available if required. Therefore, this change does not involve a significant reduction in the margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L14" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change reduces the Frequency for performance of a COT on two reactor trip Functions from 14 days to 92 days. This change is consistent with WCAP-10271-P-A, Supplement 2, Rev.1, and the 14 day current Technical Specifications Frequency is adequately bounded by the analysis of the Frequency of 31 days. The Surveillance Frequency is not assumed to be an initiator of any accident previously evaluated. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. The Surveillance Frequency does not affect the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change extends the Frequency of surveillance for the performance of a COT on the reactor functions from 14 days to 92 days. The extended time is justified by calculation for a 92 day Frequency in accordance with the company setpoint methodology procedure. The new Frequency is consistent with the WCAP 10271-P-A Surveillance Frequency for this function. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L15" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set

forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change reduces the Frequency for performance of a COT on Nuclear Instrumentation System channels from 7 days to 92 days. This change is consistent with WCAP-10271-P-A, Supplement 2, Rev.1, and the 7 day current Technical Specifications Frequency is adequately bounded by the analysis of the 31 day Frequency. The Surveillance Frequency is not assumed to be an initiator of any accident previously evaluated. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. The Surveillance Frequency does not affect the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change extends the Frequency of surveillance for the performance of a COT on Nuclear Instrumentation System channels from 7 days to 92 days. The extended time is justified by calculation for a 92 day Frequency in accordance with the company setpoint methodology procedure. The new Frequency is consistent with the WCAP 10271-P-A Surveillance Frequency for this function. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L16" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change reduces the Frequency for performance of a COT and TADOT on certain reactor trip Functions from 31 days to 92 days. This change is consistent with WCAP-10271-P-A, Supplement 2, Rev.1. The Surveillance Frequency is not assumed to be an initiator of any accident previously evaluated. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. The Surveillance Frequency does not affect the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change extends the Frequency of surveillance for the performance of a COT and TADOT on certain reactor trip functions from 31 days to 92 days. The extended time is justified by calculation for a 92 day Frequency in accordance with the company setpoint methodology procedure. The new Frequency is consistent with the WCAP 10271-P-A surveillance Frequency for this function. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L17" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change reduces the Frequency for performance of a COT on the Turbine Impulse Pressure reactor trip Function from 31 days to 18 months. The Turbine Impulse Pressure reactor trip Function is used only during unit startup to feed the P-7 permissive interlock, which bypasses other trip Functions below 10 % RTP,

and is an infrequent operation. The Surveillance Frequency is not assumed to be an initiator of any accident previously evaluated. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. The Surveillance Frequency does not affect the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. This change increases the Surveillance Frequency interval, which increases slightly the risk that a failure in the system would remain undetected between performance of surveillance tests. Thus, this change reduces the implied margin of safety associated with verifying OPERABILITY by Surveillance. However, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L18" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change reduces the Frequency of performance of an ACTUATION LOGIC TEST on Automatic Actuation Logic from 7 days to 62 days (31 days on a STAGGERED TEST BASIS). The Surveillance Frequency is not assumed to be an initiator of any accident previously evaluated. This change is based on industry operating experience, and considers instrument reliability and operating history data. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. The Surveillance Frequency does not affect the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. This change increases the Surveillance Frequency interval, which increases slightly the risk that a failure in the system would remain undetected between performance of surveillance tests. Thus, this change reduces the implied margin of safety associated with verifying OPERABILITY by Surveillance. However, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L19" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change reduces the Frequency of performance of an TADOT on the Reactor Trip Breakers from 31 days to 62 days (31 days on a STAGGERED TEST BASIS). The Surveillance Frequency is not assumed to be an initiator of any accident previously evaluated. This change is based on industry operating experience, and considers instrument reliability and operating history data. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal

plant operation, or methods of operation. This change does not introduce any new modes of operation. The Surveillance Frequency does not affect the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. This change increases the Surveillance Frequency interval, which increases slightly the risk that a failure in the system would remain undetected between performance of surveillance tests. Thus, this change reduces the implied margin of safety associated with verifying OPERABILITY by Surveillance. However, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L20" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change reduces the Frequency of performance of an TADOT on the Reactor Trip Bypass Breakers from 31 days to prior to placing the bypass breaker in service. Since the bypass breakers are only placed in service when the Reactor Trip Breakers (RTBs) are being tested, this Frequency is 62 days (31 days on a STAGGERED TEST BASIS). The Surveillance Frequency is not assumed to be an initiator of any accident previously evaluated. This change is based on industry operating experience, and considers instrument reliability and operating history data. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. The Surveillance Frequency does not affect the possibility of a new or different kind of accident from any accident

previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. This change increases the Surveillance Frequency interval, which increases slightly the risk that a failure in the system would remain undetected between performance of surveillance tests. Thus, this change reduces the implied margin of safety associated with verifying OPERABILITY by Surveillance. However, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L21" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change increases the allowed time to either restore an inoperable reactor trip Function channel to OPERABLE status, or place it in trip, from 1 hour to 6 hours. This change is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1. This condition applies to Functions that operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. Placing the inoperable channel in trip configures the Function in a one-out-of-two logic arrangement that satisfies redundancy requirements. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. The allowed time to place an inoperable channel in trip does not affect the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. This change is consistent with WCAP-10271-P-A for this function. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L22" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change increases the allowed time to restore an inoperable containment high range monitoring channel to OPERABLE status from 7 days to 30 days. The high range containment monitor is not assumed to be an initiator of any accident previously evaluated. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. The allowed time to restore an inoperable channel to OPERABLE status does not affect the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. This change extends the period of time that an inoperable channel may be out of service, which decreases slightly the implied margin of safety associated with dependence on the remaining

OPERABLE channel(s) for a longer period of time. However, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L23" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change increases the allowed time to restore an inoperable Auxiliary Feedwater (AFW) flow indicator to OPERABLE status from 7 days to 30 days. The AFW flow indicator is not assumed to be an initiator of any accident previously evaluated. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. The allowed time to restore an inoperable AFW flow indicator to OPERABLE status does not affect the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. This change extends the period of time that an inoperable channel may be out of service, which decreases slightly the implied margin of safety associated with dependence on the remaining OPERABLE channel(s) for a longer period of time. However, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L24" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change permits the unit to be placed in MODE 4 within 12 hours, rather than ≤ 200 °F within 36 hours, in the event an inoperable hydrogen monitor cannot be restored to OPERABLE status within the specified time. The hydrogen monitor is not assumed to be an initiator of any accident previously evaluated, nor will its inoperability have any impact on the probability or consequences of any accident previously evaluated. Placing the unit in MODE 4 sufficiently reduces the thermal energy to a plant condition that is well bounded by the 10 CFR 50.46 analyses, thereby significantly reducing the potential for a Loss of Coolant Accident (LOCA) that would result in a metal-water reaction. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. The inoperability of a hydrogen monitor does not impact the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L25" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change increases the time allowed to restore an inoperable incore thermocouple to OPERABLE status from 7 days to 30 days; and requires a report be prepared, rather than a unit shutdown in the event the Completion Times are not met. The incore thermocouples are not assumed to be initiators of any accident previously evaluated, nor will their inoperability have any impact on the probability or consequences of any accident previously evaluated. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. The inoperability of an incore thermocouple does not impact the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. This change extends the period of time that an inoperable channel may be out of service, which decreases slightly the implied margin of safety associated with dependence on the remaining OPERABLE channel(s) for a longer period of time. However, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L26" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change increases the time allowed to restore at least one inoperable incore thermocouple to OPERABLE status from 48 hours to 7 days. The incore thermocouples are not assumed to be initiators of any accident previously evaluated, nor will their inoperability have any impact on the probability or consequences of any accident previously evaluated. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. The inoperability of an incore thermocouple does not impact the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. This change extends the period of time that an inoperable channel may be out of service, which decreases slightly the implied margin of safety associated with dependence on the remaining OPERABLE channel(s) for a longer period of time. However, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L27" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant

hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change eliminates the requirement that a "Test" be performed on certain post accident monitoring Functions at an "R" Frequency. Surveillance testing is not assumed to be an initiator of any accident previously evaluated. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. The performance of a Surveillance test does not impact the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
<u>("L28" Labeled Comments/Discussions)</u>

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change allows an

inoperable 480V Loss of Voltage channel to be placed in block in 6 hours, rather than 1 hour, restoring to OPERABLE status in 48 hours, or shutting down the unit. This change is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1. Placing the inoperable channel in block maintains the emergency bus trip Function, because the two Loss of Voltage channels per bus are configured in a one-out-of-two logic, such that if either channel sees loss of voltage, it will trip the bus. With one channel placed in block, the OPERABLE channel is still available to trip the bus. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. This change is consistent with WCAP-10271-P-A for this Function. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L29" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change allows an inoperable 480V Degraded Voltage channel to be placed in trip in 6 hours, rather than 1 hour, restoring to OPERABLE status in 48 hours, or shutting down the unit. This change is consistent with WCAP-10271-P-A, Supplement 2, Rev. 1. Placing the inoperable channel in trip maintains the emergency bus trip Function, because the three Degraded Voltage channels per bus are

configured in a two-out-of-three logic, such that if any two channels see a degraded voltage condition, they will trip the bus. With one channel placed in trip, the two OPERABLE channels are still available to trip the bus in a one-out-of-two logic arrangement. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. This change is consistent with WCAP-10271-P-A for this Function. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L30" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The change permits an inoperable Degraded Voltage Function channel to be bypassed for up to 4 hours for surveillance testing of other channels. There are three Degraded Voltage channels per bus, and this allowance is made where bypassing the channel does not cause an actuation, and where at least two other channels per bus are monitoring the parameter. The Degraded Voltage Function is arranged in a two-out-of-three configuration. Bypassing one channel would still provide a two-out-of-two configuration. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L31" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change eliminates an OPERABILITY test of the radiation monitors which actuate containment ventilation isolation just prior to refueling operations. Since the OPERABILITY of these radiation monitoring channels is adequately verified by a CHANNEL OPERATIONAL TEST at a 92 day Frequency, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. This change extends the period of time that an inoperable channel may be out of service, which decreases slightly the implied margin of safety associated with dependence on the remaining OPERABLE channel(s) for a longer period of time. However, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L32" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change allows an inoperable Steam Generator (SG) water level AFW actuation channel to be placed in trip, rather than requiring a shutdown of the unit. Placing the inoperable channel in trip maintains the AFW pump autostart Function OPERABLE in a one-out-of-two logic configuration, instead of two-out-of-three. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L33" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change allows an inoperable 4kV undervoltage relay to be placed in trip, rather than requiring a shutdown of the unit. Placing the inoperable channel in trip maintains the AFW pump autostart Function OPERABLE in a one-out-of-one logic configuration, instead of two-out-of-two. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L34" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change reduces the MODE of Applicability for the containment ventilation isolation function to during CORE ALTERATIONS and during movement of irradiated fuel within containment. The containment ventilation isolation function is not assumed to be the initiator of any accident, nor is it taken credit for in any accident analyses other than the fuel handling accident. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L35" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change reduces the MODE of Applicability for certain functions of the reactor protection system during MODEs 3, 4, and 5. The remaining Applicability for these functions ensures that these Functions will be available to shut down the reactor when required. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L36" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change adds a Required Action to increase THERMAL POWER above the P-10 setpoint to exit the Applicability of the intermediate range instrumentation. Although an increase in THERMAL POWER is allowed, increasing power provides equivalent action to a reduction in THERMAL POWER to below the P-6 interlock within the same time period. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L37" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The reactor coolant pump breaker position trip Function is not assumed to be an initiator of accidents. The most likely event requiring the reactor coolant pump breaker position trip Function is a loss of offsite power, and the remaining trip Functions on the other two pumps provide adequate protection during the allowed outage time of 6 hours. Reducing power to below the P-8 setpoint puts the reactor in a condition where the reactor coolant pump trip Function is no longer necessary to anticipate the low Reactor Coolant System flow trip to protect the core. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change does not introduce any new modes of operation. Inoperable instrument channels cannot initiate a new or different kind of accident. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The allowed outage time of 6 hours is consistent with WCAP-P-A, Supplement 2, Rev. 1, June 1990. Reduction of THERMAL POWER to below the P-8 setpoint puts the reactor in a condition where the reactor coolant pump trip Function is no longer

necessary to anticipate the low Reactor Coolant System flow trip to protect the core Therefore, this change does not involve a significant reduction in a margin of safety.