



BROWNS FERRY NUCLEAR PLANT

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS

Enclosure I
Volume 1

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT
APPLICATION OF SCREENING CRITERIA

CONTENTS

1.	INTRODUCTION	1
2.	SCREENING CRITERIA	1
3.	PROBABILISTIC RISK ASSESSMENT INSIGHTS	4
4.	RESULTS OF APPLICATION OF SCREENING CRITERIA	8
5.	REFERENCES	8

ATTACHMENT

SUMMARY DISPOSITION MATRIX FOR BFN UNITS 1, 2, AND 3

APPENDICES

- A. JUSTIFICATION FOR SPECIFICATION RELOCATION
- B. BFN SPECIFIC RISK SIGNIFICANT EVALUATION

BROWNS FERRY NUCLEAR PLANT APPLICATION OF SCREENING CRITERIA

1.0 INTRODUCTION

The purpose of this document is to confirm the results of the BWR Owners Group application of the Technical Specification screening criteria on a plant specific basis for Browns Ferry (BFN) Units 1, 2 & 3. TVA has reviewed the application of the screening criteria to each of the Technical Specifications utilized in BWROG report NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," (Reference 1) including Supplement 1 (Reference 1), NUREG 1433, Revision 1, Standard Technical Specifications, General Electric Plants BWR/4," and applied the criteria to each of the current BFN Units 1, 2 & 3 Technical Specifications. Additionally, in accordance with the NRC guidance, this confirmation of the application of screening criteria to BFN Units 1, 2, & 3 includes confirming the risk insights from Probabilistic Risk Assessment (PRA) evaluations, provided in the Reference 1, as applicable to BFN Units 1, 2, & 3.

2. SCREENING CRITERIA

TVA used the screening criteria provided in the NRC Final Policy Statement on Technical Specification Improvements of July 22, 1993 to develop the results contained in the attached matrix. Probabilistic Risk Assessment (PRA) insights as used in the BWROG submittal were used, confirmed by TVA, and are discussed in the next section of this report. The screening criteria and discussion provided in the NRC Final Policy statement are as follows:

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

Discussion of Criterion 1: A basic concept in the adequate protection of the public health and safety is the prevention of accidents. Instrumentation is installed to detect significant abnormal degradation of the reactor coolant pressure boundary so as to allow operator actions to either correct the condition or to shut down the plant safely, thus reducing the likelihood of a loss-of-coolant accident. This criterion is intended to ensure that Technical Specifications control those instruments specifically installed to detect excessive reactor coolant system leakage. 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).

**BROWNS FERRY NUCLEAR PLANT
APPLICATION OF SCREENING CRITERIA**

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

Discussion of Criterion 2: Another basic concept in the adequate protection of the public health and safety is that the plant shall be operated within the bounds of the initial conditions assumed in the existing DBA and transient analyses and that the plant will be operated to preclude unanalyzed transients and accidents. These analyses consist of postulated events, analyzed in the 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 DBA 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 DBA 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 DBA 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 DBA 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 DBA or Transient should occur, structures, systems, and components are available to function or to actuate in order to mitigate the consequences of the DBA or Transient. Safety sequence analyses or their equivalent have been performed in recent years and provide a method of

BROWNS FERRY NUCLEAR PLANT
APPLICATION OF SCREENING CRITERIA

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 DBA 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's response to DBAs 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 DBA 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

BROWNS FERRY NUCLEAR PLANT
APPLICATION OF SCREENING CRITERIA

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 submittal. Further, as a part of the Commissions 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

The Final Policy Statement includes a statement that NRC expects licensees to utilize the available literature on risk insights to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk.

Those Technical Specifications proposed for relocation to other plant controlled documents will be maintained under the 10 CFR 50.59, safety evaluation review program. These 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) 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 Technical Specifications proposed to remain part of the Improved Technical Specifications were not reviewed. This review was accomplished in Reference 1 except where discussed in Appendix A, "Justification For Specification Relocation," and has been confirmed by TVA for those Specifications to be relocated. Where Reference 1 did not review a Technical Specification against the criteria of Reference 3, TVA performed a review similar (but not identical) to that described below for Reference 1. The results of these reviews are presented in Appendix B.

**BROWNS FERRY NUCLEAR PLANT
APPLICATION OF SCREENING CRITERIA**

Assumptions and Approach

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

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

- a. List the function(s) affected by removal of the LCO item.
- b. Determine the effect of loss of the LCO item on the function(s).
- c. Identify compensating provisions, redundancy, and backups related to the loss of the LCO item.
- d. Determine the relative frequency (high, medium, and low) of the loss of the function(s) assuming the LCO item is removed from Technical Specifications and controlled by other procedures or programs. Use information from current PRAs and related analyses to establish the relative frequency.
- e. Determine the relative significance (high, medium, and low) of the loss of the function(s). Use information from current PRAs and related analyses to establish the relative significance.

**BROWNS FERRY NUCLEAR PLANT
APPLICATION OF SCREENING CRITERIA**

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

RISK CRITERIA

Consequence

<u>Frequency</u>	<u>High</u>	<u>Medium</u>	<u>Low</u>
High	S	S	NS
Medium	S	S	NS
Low	NS	NS	NS

S = Potential Significant Risk Contributor
 NS = Risk Non-Significant

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

BROWNS FERRY NUCLEAR PLANT
APPLICATION OF SCREENING CRITERIA

TABLE 3-1

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

- BWR/6 Standard Plant, GESSAR II, 238 Nuclear Island, BWR/5 Standard Plant Probabilistic Risk Assessment, Docket No. STN 50- 447, March 1982.
- La Salle County Station, NEDO-31085, Probabilistic Safety Analysis, February 1988.
- Grand Gulf Nuclear Station, IDCOR, Technical Report 86.2GG, Verification of IPE for Grand Gulf, March 1987.
- Limerick, Docket Nos. 50-352, 50-353, 1981, "Probabilistic Risk Assessment, Limerick Generating Station," Philadelphia Electric Company.
- Shoreham, Probabilistic Risk Assessment Shoreham Nuclear Power Station, Long Island Lighting Company, SAI-372-83-PA-01, June 24, 1983.
- Peach Bottom 2, NUREG-75/0104, "Reactor Safety Study," WASH-1400, October 1975.
- Millstone Point 1, NUREG/CR-3085, "Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 1 Nuclear Power Plant," January 1983.
- Grand Gulf, NUREG/CR-1559, "Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant," October 1981.
- NEDC-30935P, "BWR Owners' Group Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation) Part 2," June 1987.

**BROWNS FERRY NUCLEAR PLANT
APPLICATION OF SCREENING CRITERIA**

4. RESULTS OF APPLICATION OF SCREENING CRITERIA

The screening criteria from Section 2 were applied to the BFN Units 1, 2, and 3 Technical Specifications. The attachment is a summary of that application indicating which Specifications are being retained or relocated. Discussions that document the rationale for the relocation of each Specification which failed to meet the screening criteria are provided in Appendix A. No Significant Hazards Considerations (10 CFR 50.92) evaluations for those Specifications relocated are provided with the Justification for Changes for the specific Technical Specifications. TVA will relocate those Specifications identified as not satisfying the criteria to licensee controlled documents whose changes are governed by 10 CFR 50.59.

5. REFERENCES

1. NEDO-31466 (and Supplement 1), "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.
2. NUREG 1433, Revision 1, "Standard Technical Specifications, General Electric Plants BWR/4," April 1995.
3. Final Policy Statement on Technical Specifications Improvements, July 22, 1993, (58FR39132).

SUMMARY DISPOSITION MATRIX FOR BFN UNITS 1, 2, AND 3

CURRENT TS NUMBER	TITLE	BFN ISTS NUMBER	STS REV. 4 NUMBER	NUREG 1433 NUMBER	RETAINED/ CRITERION FOR INCLUSION	BASIS FOR INCLUSION/EXCLUSION ^(a)
1.0	Definitions	1.1	1.0	1.1	Yes	Definitions for selected terms used in the Technical Specifications are provided to improve understanding and ensure consistent application. Application of the Technical Specification selection criteria to these definitions is not appropriate. However, definitions for those terms that remain in the Technical Specifications following the application of the selection criteria will be retained.
1/2.1.A 1/2.1.B 1/2.1.C	Safety Limit: Fuel Cladding Integrity	2.1 3.3.1.1 3.3.5.1 3.3.5.2 3.3.6.1	2.1.1 2.1.2 2.1.4 2.2.1 3.3.2 3.3.3 3.3.5	2.1 3.3.1.1 3.3.5.1 3.3.6.1	Yes	Application of Technical Specification selection criteria to Safety Limits and Limiting Safety System Settings (LSSS) is not appropriate. The fuel cladding integrity LSSS (with the exception of APRM Rod Blocks) are retained by their incorporation into the RPS and ECCS instrumentation Specifications because the associated Functions either actuate to mitigate consequences of Design Basis Accidents (DBAs) and transients or are retained as directed by the NRC.
2.1.A.1.c	Safety Limit: Fuel Cladding Integrity -- APRM Rod Block Trip Setting	Relocated	None	None	No	See Appendix A, page A-4.
1/2.2	Safety Limit: Reactor Coolant System Integrity	2.1.2 3.3.1.1 3.4.3 3.3.6.1	2.1.3 2.2 3.4.2.1	2.1.2 3.3.1.1 3.4.3 3.3.6.1	Yes	Application of Technical Specification selection criteria to Safety Limits and Limiting Safety System Settings (LSSS) is not appropriate. The Reactor Coolant System integrity LSSS are retained by incorporation into the RPS and safety relief valve Specifications because the associated components function to mitigate the consequences of events that would result in overpressurization of the RCS.
1.0.C	Limiting Condition for Operation (LCO) Applicability	LCO 3.0.3 3.8.1	3.0.3 3/4.8.1.1	LCO 3.0.3 3.8.1	Yes	This Specification provides generic guidance applicable to one or more Specifications to facilitate understanding of LCOs. As such, direct application of the Technical Specification selection criteria is not appropriate. The general requirements of 1.0.C are retained in the Technical Specifications consistent with NUREG-1433.

SUMMARY DISPOSITION MATRIX FOR BFN UNITS 1, 2, AND 3

CURRENT TS NUMBER	TITLE	BFN ISTS NUMBER	STS REV. 4 NUMBER	NUREG 1433 NUMBER	RETAINED/ CRITERION FOR INCLUSION	BASIS FOR INCLUSION/EXCLUSION ^(a)
1.0.LL	Surveillance Requirement (SR) Applicability	SR 3.0.1 SR 3.0.2 SR 3.0.3	4.0.1 4.0.2 4.0.3	SR 3.0.1 SR 3.0.2 SR 3.0.3	Yes	This Specification provides generic guidance applicable to one or more Specifications to facilitate understanding of SRs. As such, direct application of the Technical Specification selection criteria is not appropriate. The general requirements of 1.0.LL are retained in the Technical Specifications consistent with NUREG-1433.
3/4.1.A	Reactor Protection System: Instrumentation that Initiate a Reactor Scram (Instruments in Table 3.1.1 and associated SRs in Table 4.1.1 and 4.1.2)	3.3.1.1	3/4.3.1	3.3.1.1	Yes - 3, 4	All Functions retained (with exception listed below) because the various Functions: 1) actuate to mitigate consequences of DBAs and/or transients; or, 2) are considered risk significant and retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements; or, 3) are part of the RPS/Reactor Scram Function; or, 4) provide an anticipatory scram to ensure the scram discharge volume and thus RPS remains operable.
Table 3/4.1.A	Turbine First Stage Permissive	Relocated	None	None	No	See RPS Instrumentation Justification for Change (LA5 for ISTS 3.3.1.1).
3/4.1.B	Reactor Protection System Power Supply	3.3.8.2	3/4.8.4.4	3.3.8.2	Yes - 3	Provides protection for the RPS bus powered instrumentation against unacceptable voltage and frequency conditions that could degrade instrumentation so that it would not perform the intended safety function.
3/4.2.A	Primary Containment and Reactor Building Isolation: Instrumentation that initiates primary containment isolation. (Instruments in Table 3.2.A and associated SRs in Table 4.2.A) (Exceptions listed below)	3.3.6.1 3.3.6.2 3.3.7.1	3/4.3.2	3.3.6.1 3.3.6.2 3.3.7.1	Yes - 3, 4	All Functions retained (with exceptions listed below) because the Functions actuate to mitigate the consequences of a DBA LOCA, Fuel Handling Accident or are considered risk significant and are retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements. The isolation signals generated by the reactor building isolation instrumentation are implicitly assumed in the safety analyses to initiate closure of valves to limit offsite doses.
3/4.2.A	SGTS flow functions	Relocated	None	None	No	See Secondary Containment Isolation Instrumentation Justification for Change (LA3 for ISTS 3.3.6.2)
3/4.2.A	Reactor Building Isolation Timer Functions	Relocated	None	None	No	See Secondary Containment Isolation Instrumentation Justification for Change (LA3 for ISTS 3.3.6.2)

SUMMARY DISPOSITION MATRIX FOR BFN UNITS 1, 2, AND 3

CURRENT TS NUMBER	TITLE	BFN ISTS NUMBER	STS REV. 4 NUMBER	NUREG 1433 NUMBER	RETAINED/ CRITERION FOR INCLUSION	BASIS FOR INCLUSION/EXCLUSION ^(a)
3/4.2.B	Core and Containment Cooling Systems - Initiation & Control: Instrumentation that initiates or controls the core and containment cooling systems (LPCI, CS, ADS, HPCI, and RCIC). (Instruments in Table 3.2.B and associated SRs in Table 4.2.B) (Exceptions listed below)	3.3.5.1 3.3.5.2 3.3.6.1	3/4.3.3 3/4.3.5	3.3.5.1 3.3.5.2 3.3.6.1	Yes - 3, 4	Functions retained (with exceptions listed below) because the various Functions actuate to mitigate the consequences of a DBA LOCA or are considered risk significant and are retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements.
Table 3/4.2.B	Drywell High Pressure (1<ps2.5 psig)	Relocated	None	None	No	See ECCS Instrumentation Justification for Change (R2 for ISTS 3.3.5.1)
Table 3/4.2.B	Core Spray Sparger to Reactor Pressure Vessel d/p	Relocated	None	None	No	See Appendix A, Page A-3.
Table 3/4.2.B	Trip System Bus Power Monitors: RHR (LPCI), CS, ADS, HPCI, and RCIC Trip Systems	Relocated	None	None	No	See Appendix A, Page A-1
Table 3/4.2.B	CS and RHR Discharge Pressure	Relocated	None	None	No	See ECCS Instrumentation Justification for Change (LA3 for ISTS 3.3.5.1)
Table 3/4.2.B	End of Cycle Recirculation Pump Trip: Instrumentation that trips the reactor recirculation pump to limit the consequences of a failure to scram (ATWS-RPT). (Instruments in Table 3.2.B and associated SRs in Table 4.2.B))	3.3.4.1	3/4.3.4.1	3.3.4.1	Yes - 3	ECC-RPT aids the reactor scram in protecting fuel cladding integrity by ensuring the fuel cladding integrity Safety Limit is not exceeded during a load rejection or turbine trip transient.
Table 3/4.2.B	CS and RHR Area Cooler Fan Thermostat	Relocated	None	None	No	See ECCS Instrumentation Justification for Change (LA3 for ISTS 3.3.5.1)
3/4.2.C	Control Rod Block Actuation: Instrumentation that Initiates Control Rod Blocks. (Instruments in Table 3.2.C and associated SRs in Table 4.2.C) (Exceptions listed below)	3.3.2.1	3/4.3.6	3.3.2.1	Yes	Control Rod Block Actuation Instrumentation functions to prevent violation of the MCPR Safety Limit and cladding plastic strain design limit during a single control rod withdrawal error event, ensures the initial conditions of the control rod drop accident analysis are not violated, and prevents inadvertent criticality when the reactor is shutdown (thereby preserving the safety analysis assumptions).

SUMMARY DISPOSITION MATRIX FOR BFN UNITS 1, 2, AND 3

CURRENT TS NUMBER	TITLE	BFN ISTS NUMBER	STS REV. 4 NUMBER	NUREG 1433 NUMBER	RETAINED/ CRITERION FOR INCLUSION	BASIS FOR INCLUSION/EXCLUSION ^(a)
Table 3/4.2.C	APRM (Upscale Flow Bias, Upscale Startup Mode, APRM Downscale, and APRM Inoperative)	Relocated	3/4.3.6	None	No	See Appendix A, Page A-4.
Table 3/4.2.C	IRM (Upscale, Downscale, Detector Not in Startup Position, Inoperative)	Relocated	3/4.3.6	None	No	See Appendix A, Page A-6.
Table 3/4.2.C	SRM (Upscale, Downscale, Detector Not in Startup Position, Inoperative)	Relocated	3/4.3.6	None	No	See Appendix A, Page A-7.
Table 3/4.2.C	Scram Discharge Volume High Level	Relocated	3/4.3.6	None	No	See Appendix A, Page A-8.
3/4.2.D	(Deleted)					
3/4.2.E	Drywell Leak Detection: Instrumentation that monitors drywell leakage	3.4.5	3/4.4.3.1	3.4.6	Yes - 1	Leak detection instrumentation is used to indicate an abnormal condition of the reactor coolant pressure boundary.
3/4.2.F 3/4.7.H	Surveillance Instrumentation (Post Accident Monitoring Instruments): Instrumentation that provide surveillance information. (Instruments in Table 3.2.F and associated SRs in Table 4.2.F)	3.3.3.1	3/4.3.7.5	3.3.3.1	Yes - 3	Regulatory Guide 1.97 Type A and Category 1 instruments retained. See Appendix A, Page A-10, for full discussion of all instruments in Table 3.2.F.
3/4.2.G	Control Room Isolation: Instrumentation that isolates the control room and initiates CREVs. (Instruments in Table 3.2.G and associated SRs in Table 4.2.G)	3.3.7.1	3/4.3.7	3.3.7.1	Yes - 3	Functions actuate to maintain control room habitability so that operation can continue from the control room following a DBA.
3/4.2.H	Flood Protection Instrumentation	Relocated	None	None	No	See Appendix A, page A-16.
3/4.2.I	Meteorological Monitoring Instrumentation	Relocated	None	None	No	See Appendix A, page A-19.
3/4.2.J	Seismic Monitoring Instrumentation	Relocated	None	None	No	See Appendix A, page A-18.

SUMMARY DISPOSITION MATRIX FOR BFN UNITS 1, 2, AND 3

CURRENT TS NUMBER	TITLE	BFN ISTS NUMBER	STS REV. 4 NUMBER	NUREG 1433 NUMBER	RETAINED/ CRITERION FOR INCLUSION	BASIS FOR INCLUSION/EXCLUSION ^(a)
3/4.2.K	Explosive Gas Monitoring Instrumentation	Relocated	None	None	No	See Appendix A, page A-21. Part of the program required by BFN Specification 5.5.8.
3/4.2.L	ATWS Recirculation Pump Trip	3.3.4.2	3/4.3.4.1	3.3.4.2	Yes - 4	ATWS-RPT is being retained in accordance with NRC Final Policy Statement on Technical Specification Improvements due to risk significance.
3/4.3.A.1	Reactivity Margin - Core Loading	3.1.1	3/4.1.1	3.1.1	Yes - 2	Shutdown Margin (SDM) is assumed as an initial condition for the control rod removal error during a refueling event and the fuel assembly insertion error during a refueling event.
3/4.3.A.2.a 3/4.3.A.2.b 3/4.3.A.2.c 3.3.A.2.d 3/4.3.B.1	Reactivity Margin - Inoperable Control Rods	3.1.3	3/4.1.3.1 3/4.1.3.5 3/4.1.3.6	3.1.3	Yes - 3	Control rods are part of the primary success path for mitigating the consequences of DBAs and transients.
3.3.A.2.e /4.3.A.2.d	Scram Accumulators	3.1.5	3/4.1.3.5 3/4.1.3.7	3.1.5	Yes - 3	Same as above.
3/4.3.B.2	Control Rod Housing Support	Relocated	3/4.1.3.8	None	No	See Control Rod Operability Justification for Change (R1 for ISTS 3.1.3)
3/4.3.B.3.b	Rod Worth Minimizer	3.3.2.1	3/4.1.4.1	3.3.2.1	Yes - 3	The RWM enforces the Banked Position Withdrawal Sequence (BPWS) to ensure that the initial conditions of the LOCA analysis are not violated.
3/4.3.B.4	Minimum Count Rate for Control Rod Withdrawal	3.3.1.2	3/4.3.7.6	3.3.1.2	Yes	Does not satisfy selection criteria, however is being retained because it is considered necessary for flux monitoring during shutdown, startup and refueling operations.
3/4.3.C	Scram Insertion Times	3.1.3 3.1.4	3/4.1.3.2 3/4.1.3.3 3/4.1.3.4	3.1.3 3.1.4	Yes - 3	Control rods are part of the primary success path for mitigating the consequences of DBAs and transients. The LOCA and transient analyses assume that control rods scram at a specified insertion rate.
3/4.3.D	Reactivity Anomalies	3.1.2	3/4.1.2	3.1.2	Yes - 2	Not a measured process variable, but is important parameter that is used to confirm the acceptability of the accident analysis.
3/4.3.F	Scram Discharge Volume	3.1.8	3/4.1.3.1	3.1.8	Yes - 3	The capability to insert the control rods ensures the assumptions used for the scram reactivity in the LOCA and transient analyses are maintained. The Scram Discharge Volume (SDV vent and drain valves contribute to the operability of the control rod scram function.

SUMMARY DISPOSITION MATRIX FOR BFN UNITS 1, 2, AND 3

CURRENT TS NUMBER	TITLE	BFN ISTS NUMBER	STS REV. 4 NUMBER	NUREG 1433 NUMBER	RETAINED/ CRITERION FOR INCLUSION	BASIS FOR INCLUSION/EXCLUSION ^(a)
3/4.4	Standby Liquid Control System	3.1.7	3/4.1.5	3.1.7	Yes - 4	The Standby Liquid Control (SLC) is a backup system to the control rod scram function. This system is being retained per the NRC Final Policy Statement on Technical Specification Improvements due to the risk significance.
	CORE AND CONTAINMENT COOLING SYSTEMS					
3/4.5.A	Core Spray System	3.5.1	3/4.5.1	3.5.1	Yes - 3	Core Spray subsystems are part of the ECCS and function to provide cooling water to the reactor core to mitigate large Loss of Coolant Accidents.
3/4.5.B	Residual Heat Removal System (RHR) (LPCI AND Containment Cooling)	3.5.1 3.5.2 3.6.2.3 3.6.2.4 3.6.2.5	3/4.5.1 3/4.5.2 3/4.6.2.2 3/4.6.2.3	3.5.1 3.5.2 3.6.2.3 3.6.2.4 3.6.2.5	Yes - 3	RHR Low Pressure Coolant Injection subsystems are part of the ECCS and function to provide cooling water to the reactor core to mitigate large Loss of Coolant Accidents. RHR Containment Cooling systems provide a reliable source of cooling water and functions to provide cooling to the primary containment under post accident conditions.
3/4.5.B.11 3/4.5.B.12 3/4.5.B.13	RHR cross-connect capability between units	None	None	None	Relocated	See Justification for Change R1 for BFN ISTS 3.5.1, ECCS.
3/4.5.C.1 3/4.5.C.2	RHR Service Water and Emergency Equipment Cooling Water Systems	3.7.1 3.7.2	3/4.7.1.1 3/4.7.1.2 3/4.7.1.3	3.7.1 3.7.2	Yes - 3	Designed for heat removal from various safety related systems following a DBA. As such, acts to mitigate the consequences of an accident.
3/4.5.C.3 3/4.5.C.4 3/4.5.C.5	Standby Coolant Supply Capability	None	None	None	Relocated	See Justification for Change R1 for BFN ISTS 3.7.1, RHRSW.
3/4.5D	Equipment Area Coolers	Relocated	None	None	No	Relocated to the Bases as they are part of ECCS Operability. See ECCS Justification for Changes (3.5.1, LA4)
3/4.5E	High Pressure Coolant Injection System	3.5.1	3/4.5.1	3.5.1	Yes - 3	The HPCI System is part of the ECCS and functions to mitigate small break Loss of Coolant Accidents.
3/4.5F	Reactor Core Isolation Cooling System	3.5.3	3/4.7.4	3.5.3	Yes - 4	System retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements due to risk significance.

SUMMARY DISPOSITION MATRIX FOR BFN UNITS 1, 2, AND 3

CURRENT TS NUMBER	TITLE	BFN ISTS NUMBER	STS REV. 4 NUMBER	NUREG 1433 NUMBER	RETAINED/ CRITERION FOR INCLUSION	BASIS FOR INCLUSION/EXCLUSION ^(a)
3/4.5G	Automatic Depressurization System (ADS)	3.5.1	3/4.5.1	3.5.1	Yes - 3	The ADS is part of the ECCS and is designed to mitigate a small or medium break Loss of Coolant Accident. The ADS acts to rapidly reduce reactor vessel pressure in a LOCA situation in which the HPCI System fails to automatically maintain reactor vessel water level. This depressurization enables the low-pressure emergency core cooling systems to deliver cooling water to the reactor core.
3/4.5H	Maintenance of Filled Discharge Pipe	3.5.1 3.5.2 3.5.3	3/4.5.1 3/4.5.2 3/4.5.4	3.5.1 3.5.2 3.5.3	Yes - 3, 4	This Specification ensures the operability of the ECCS and RCIC System, which function to mitigate the consequences of a LOCA (ECCS) or is required to be retained by the NRC Final Policy Statement on Technical Specification Improvements (RCIC).
3/4.5I	Average Planar Linear Heat Generation Rate (MAPLHGR)	3.2.1	3/4.2.1	3.2.1	Yes - 2	The APLHGR limit is an initial condition in the safety analyses.
3/4.5J	Linear Heat Generation Rate (LHGR)	3.2.3	3/4.2.4	3.2.3	Yes - 2	The LHGR limit is an initial condition in the safety analyses.
3/4.5K	Minimum Critical Power Ratio (MCPR)	3.2.2	3/4.2.3	3.2.2	Yes - 2	The MCPR limit is an initial condition in the safety analyses.
3/4.5L	APRM Setpoints	3.2.4	3/4.2.2	3.2.4	Yes - 2, 3	The Operability of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram.
3/4.5M	Core Thermal-Hydraulic Stability	3.4.1	3/4.4.1.1 3/4.4.1.3	3.4.1	Yes - 2	Recirculation loop flow is an initial condition in the safety analysis.
3/4.6.A.1 3/4.6.A.2 3/4.6.A.3 3/4.6.A.4 3/3.6.A.5	Thermal and Pressurization Limitations	3.4.9	3/4.4.6.1	3.4.10	Yes - 2	Establishes initial conditions such that operation is prohibited in areas or at temperature rate changes that might cause undetected flaws to propagate in turn challenging the reactor coolant pressure boundary integrity.
3/4.6.A.6 3/4.6.A.7	Idle Recirculation Loop Startup	3.4.9	3/4.4.6.1	3.4.10	Yes - 2	Same as above.
3/4.6.B.1 3/4.6.B.2 3/4.6.B.3 3/4.6.B.4 3/4.6.B.5	Coolant Chemistry	Relocated	3/4.4.4	None	No	See Appendix A, Page A-12.

SUMMARY DISPOSITION MATRIX FOR BFN UNITS 1, 2, AND 3

CURRENT TS NUMBER	TITLE	BFN 1STS NUMBER	STS REV. 4 NUMBER	NUREG 1433 NUMBER	RETAINED/ CRITERION FOR INCLUSION	BASIS FOR INCLUSION/EXCLUSION ^(a)
3/4.6.B.6	Specific Activity	3.4.6	3/4.4.5	3.4.7	Yes - 2	The specific activity in the reactor coolant is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment.
3/4.6.C.1	Coolant Leakage	3.4.4	3/4.4.3.1	3.4.4	Yes - 1, 2	Leakage beyond limits would indicate an abnormal condition of the reactor coolant pressure boundary. Operation in this condition may result in reactor coolant pressure boundary failure. Leakage detection instruments are used to indicate an abnormal condition of the reactor coolant pressure boundary.
3/4.6.C.2	Leakage Detection Systems	3.4.5	3/4.4.3.2	3.4.6	Yes - 1, 2	Same as above.
3/4.6.D	Relief Valves	3.4.3	3/4.4.2.1	3.4.3	Yes - 3	The Safety and Relief Valves are assumed to operate to maintain the reactor pressure below design limits.
3/4.6E	Jet Pumps	3.4.2	3/4.4.1.2	3.4.2	Yes - 2	Jet Pump operability is explicitly assumed in the design basis LOCA to assure adequate core reflood capability.
3/4.6F	Recirculation Pump Operation	3.4.1	3/4.4.1.1 3/4.4.1.3	3.4.1	Yes - 2	Recirculation loop flow is an initial condition in the safety analysis.
3/4.6G	Structural Integrity	Relocated	3/4.4.8	None	No	See Appendix A, Page A-13
3/4.6H	Snubbers	Relocated	3/4.7.5	None	No	See Justification for Change (CTS 3.6.H/4.6.H, LA1) for relocating snubbers in CTS 3.6.H/4.6.H. (Spec 3.4 markup)
3/4.7.A.1	Suppression Chamber	3.6.2.1 3.6.2.2	3/4.6.2.1	3.6.2.1 3.6.2.2	Yes - 2, 3	The suppression pool water volume and temperature are initial conditions in the DBA LOCA containment response analysis and mitigate the consequences of a DBA.
3/4.7.A.2	Primary Containment Integrity	3.6.1.1 3.6.1.2 3.6.1.3	3/4.6.1.1 3/4.6.1.2 3/4.6.1.3 3/4.6.1.5	3.6.1.1 3.6.1.2 3.6.1.3	Yes - 3	Primary containment functions to mitigate the consequences of a DBA. Primary containment leakage is an assumption utilized in the LOCA safety analysis to ensure primary containment operability.
3/4.7.A.3	Pressure Suppression Chamber - Reactor Building Vacuum Breakers	3.6.1.5	3/4.6.4.2	3.6.1.7	Yes - 3	Pressure suppression chamber to reactor building vacuum breaker operation is relied upon to limit a negative pressure differential, secondary to primary containment, that could challenge primary containment integrity.
3/4.7.A.4	Drywell-pressure Suppression Chamber Vacuum Breakers	3.6.1.6	3/4.6.4.1	3.6.1.8	Yes - 3	Drywell-pressure suppression chamber vacuum breaker operation is assumed in the LOCA analysis to limit drywell pressure thereby ensuring primary containment integrity.

SUMMARY DISPOSITION MATRIX FOR BFN UNITS 1, 2, AND 3

CURRENT TS NUMBER	TITLE	BFN ISTS NUMBER	STS REV. 4 NUMBER	NUREG 1433 NUMBER	RETAINED/ CRITERION FOR INCLUSION	BASIS FOR INCLUSION/EXCLUSION ^(a)
3/4.7.A.5	Oxygen Concentration	3.6.3.2	3/4.6.6.4	3.6.3.2	Yes - 2	Oxygen concentration is limited such that, when combined with hydrogen (that is postulated to evolve following a LOCA), the total explosive gas concentration remains below explosive levels. Therefore, primary containment integrity is maintained.
3/4.7.A.6	Drywell-suppression Chamber Differential Pressure	3.6.2.6	3/4.6.2.4	3.6.2.5	Yes - 2	Drywell-suppression Chamber Differential Pressure is an initial condition in the DBA LOCA containment response analysis.
3/4.7B	Standby Gas Treatment System	3.6.4.3	3/4.6.5.3	3.6.4.3	Yes - 3	System functions following a DBA to limit offsite releases.
3/4.7C	Secondary Containment	3.6.4.1 3.6.4.2	3/4.6.5.1 3/4.6.5.2	3.6.4.1 3.6.4.2	Yes - 3	Secondary containment integrity is relied on to limit the offsite dose during an accident by ensuring a release to containment is delayed and treated prior to release to the environment. Damper operation within time limits establishes secondary containment and limits offsite releases to acceptable values.
3/4.7D 3.7.F.3.a	Primary Containment Isolation Valves	3.6.1.3	3/4.6.3 3/4.6.1.8	3.6.1.3	Yes - 3	Isolation valves function to limit DBA consequences.
3/4.7F	Primary Containment Purge System	Relocated	None	None	No	See Justification for Change (CTS 3.7.F/4.7.F, R1 at the end of markup for proposed BFN ISTS 3.6) for relocating primary containment purge system.
3/4.7E	Control Room Emergency Ventilation	3.7.3	3/4.7.2	3.7.4	Yes - 3	Maintains habitability of the control room so that operators can remain in the control room following an accident. As such, it mitigates the consequences of an accident by allowing the operators to continue accident mitigation activities from the control room.
3/4.7.G	Containment Atmosphere Dilution System (CAD)	3.6.3.1	3/4.6.6.2	3.6.3.4	Yes - 3	System ensures oxygen concentration is maintained below the explosive level following a LOCA by inerting the drywell with nitrogen. Therefore, containment integrity is maintained.
3.8.A.5 3/4.8.A.6	Liquid Holdup Tanks	5.5.8	None	5.5.8	Yes	Although this Specification does not meet any criteria of the NRC Final Policy Statement, it has been retained in accordance with NRC Letter from W. T. Russell to the Industry ITS chairpersons, dated October 25, 1993.

SUMMARY DISPOSITION MATRIX FOR BFN UNITS 1, 2, AND 3

CURRENT TS NUMBER	TITLE	BFN ISTS NUMBER	STS REV. 4 NUMBER	NUREG 1433 NUMBER	RETAINED/ CRITERION FOR INCLUSION	BASIS FOR INCLUSION/EXCLUSION ^(a)
3.8.B.9 3.8.B.10 4.8.B.5	Airborne Effluents - Explosive Gas Mixture	5.5.8	None	5.5.8	Yes	See Appendix A, page A-20. This is a requirement of the program required by BFN Specification 5.5.8.
3/4.8.E	Miscellaneous Radioactive Materials Sources	Relocated	3/4.7.6	None	No	See Appendix A, page A-17.
3/4.9.A.1 3.9.A.2 3.9.A.6 3/4.9.B.1 3/4.9.B.3 3.9.B.15	Auxiliary Electrical Equipment - A. C. Sources Operating	3.8.1	3/4.8.1.1	3.8.1	Yes - 3	The operability of the AC power sources is part of the primary success path of the accident analyses.
3.9.A.3 /4.9.A.4.D 4.9.A.5 3/4.9.B.2 3/4.9.B.4 3/4.9.B.5 3/4.9.B.6 3.9.B.12 3.9.B.13 3.9.B.14 3.9.B.15	Auxiliary Electrical Equipment - Buses and Boards Available	3.8.7	3/4.8.3.1	3.8.9	Yes - 3	The operability of the distribution system is part of the primary success path of the accident analyses.
3.9.A.4 /4.9.A.2 3.9.B.7 3.9.B.8 3.9.B.15	D. C. Power System	3.8.4	3/4.8.3.1	3.8.4	Yes - 3	The operability of the DC subsystems is consistent with the initial assumptions of the accident analyses.
3.9.A.5 /4.9.A.3.a	Logic Systems	3.8.1 3.3.5.1	3/4.8.1.1	3.8.1 3.3.5.1	Yes - 3	Required to mitigate the consequences of a DBA.
3/4.9.C.1 3.9.C.2	A. C. Sources - Operation in Cold Shutdown	3.8.2	3/4.8.1.2	3.8.2	Yes - 3	Same as above.
3.9.C.3 3.9.C.4	Onsite Electrical Power Distribution - Shutdown	3.8.8	3/4.8.1.2	3.8.10	Yes - 3	Same as above.
3/4.9.D	Unit 3 Diesel Generators Required for Unit 2 Operation	3.8.1 3.8.2	3/4.8.1.1 3/4.8.1.2	3.8.1 3.8.2	Yes - 3	Same as above.

SUMMARY DISPOSITION MATRIX FOR BFN UNITS 1, 2, AND 3

CURRENT TS NUMBER	TITLE	BFN ISTS NUMBER	STS REV. 4 NUMBER	NUREG 1433 NUMBER	RETAINED/ CRITERION FOR INCLUSION	BASIS FOR INCLUSION/EXCLUSION ^(a)
3/4.10.A.1 3/4.10.A.2	Refueling Operations - Interlocks	3.9.1 3.9.2 3.9.3	3/4.9.1	3.9.1 3.9.2 3.9.3	Yes - 3	The refueling interlocks protect against prompt reactivity excursions during the Refuel Mode. The safety analyses for the control rod removal error during refueling and the fuel assembly insertion error during refueling assume the functioning of the refueling interlocks.
3.10.A.3 3.10.A.4	Refueling Platform Equipment Interlocks	Relocated	3/4.9.7	None	No	See Appendix A, Page A-22.
3/4.10.A.5	Refueling Operations - Single Control Rod Maintenance	3.10.4	3/4.9.10.2	3.10.4	Yes	This requirement is being retained to allow relaxation of certain Limiting Conditions for operation (LCOs) under specific conditions to allow testing and maintenance. This requirement is directly related to several LCOs. Direct application of the Technical Specification selection criteria is not appropriate. However, this requirement, directly tied to LCOs that remain in Technical Specifications, will also remain in Technical Specifications.
3/4.10.A.6	Refueling Operations - Removal of Two Control Rods	3.10.5	3/4.9.10.2	3.10.4	Yes	Same as above.
3/4.10.A.7	Refueling Operations - Removal of Any Number of Control Rods	3.10.6	3/4.9.10.2	3.10.6	Yes	Same as above.
3/4.10.B	Refueling Operations - Core Monitoring	3.3.1.2	3/4.9.2	3.3.1.2	Yes	Does not satisfy criteria for inclusion but is retained because it is considered necessary for flux monitoring during shutdown, startup, and refueling operations.
3/4.10.D	Refueling Operations - Reactor Building Crane	Relocated	3/4.9.6	None	No	See Appendix A, Page A-14
3/4.10.E 3.10.F	Refueling Operations - Spent Fuel Cask	Relocated	3/4.9.6	None	No	See Appendix A, Page A-15
5.0	Major Design Features	4.0	5.0	4.0	Yes	Application of Technical Specification selection criteria is not appropriate. However, Design Features will be included in Technical Specifications as required by 10 CFR 50.36a.
6.0	Administrative Controls	5.0	6.0	5.0	Yes	Application of Technical Specification selection criteria is not appropriate. However, Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36a.

APPENDIX A
JUSTIFICATION FOR
SPECIFICATION RELOCATION

APPENDIX A

TABLE 3/4.2.B TRIP SYSTEM BUS POWER MONITORS FOR THE RHR (LPCI), CORE SPRAY, ADS, HPCI AND RCIC TRIP SYSTEMS

LCO Statement:

The limiting conditions for operation for the instrumentation that initiates or controls the core and containment cooling systems are given in Table 3.2.B.

Table 3.2.B Instrumentation that Initiates or Controls the Core and Containment Cooling Systems

3/4.2-17	RHR (LPCI) Trip System Bus Power Monitor
3/4.2-17	Core Spray Trip System Bus Power Monitor
3/4.2-17	ADS Trip System Bus Power Monitor
3/4.2-18	HPCI Trip System Bus Power Monitor
3/4.2-18	RCIC Trip System Bus Power Monitor

Discussion:

The Trip System Bus Power Monitors for the RHR (LPCI), Core Spray, ADS, HPCI and RCIC trip systems alarm if a fault is detected in the power system to the appropriate systems logic. No design basis accident (DBA) or transient analyses takes credit for the Trip System Bus Power Monitors. This instrumentation provides a monitoring/alarm function only.

Comparison to Screening Criteria:

1. The Trip System Bus Power Monitors are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. The Trip System Bus Power Monitors are not process variables that are initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. The Trip System Bus Power Monitors are not part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Sections 3.5 and 6 of NEDO-31466 and summarized in Table 4-1 (item 106) of NEDO-31466, Supplement 1, and verified by TVA, the loss of the RHR (LPCI), Core Spray, ADS, HPCI and RCIC Trip System Bus Power Monitors was found to be a non-significant risk contributor to core damage frequency and offsite releases.

APPENDIX A

TABLE 3/4.2.B TRIP SYSTEM BUS POWER MONITORS FOR THE RHR (LPCI), CORE
SPRAY, ADS, HPCI AND RCIC TRIP SYSTEMS (cont'd.)

Conclusion:

Since the screening criteria have not been satisfied, the RHR (LPCI), Core Spray, ADS, HPCI and RCIC Trip System Bus Power Monitors LCO and Surveillances may be relocated to a licensee controlled document.

APPENDIX A

TABLE 3/4.2.B CORE SPRAY SPARGER TO REACTOR PRESSURE VESSEL d/p

LCO Statement:

The limiting conditions for operation for the instrumentation that initiates or controls the core and containment cooling systems are given in Table 3.2.B.

Table 3.2.B Instrumentation that Initiates or Controls the Core and Containment Cooling Systems

3/4.2-17 Core Spray Sparger to Reactor Pressure Vessel d/p

Discussion:

This instrumentation measures the differential pressure between the core spray sparger and the reactor pressure vessel above the core plate and alarms if a break is detected. This Function does not actuate any equipment; it provides an alarm function only. This Function monitors the integrity of the core spray system piping in the reactor annulus region which would not otherwise be apparent to the operators. It is not credited in the accident analysis.

Comparison to Screening Criteria:

1. This instrumentation is not the primary method for detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. This instrumentation is not a process variable that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. This instrumentation is not part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Appendix B, Page 1, TVA found the loss of the Core Spray Sparger to Reactor Pressure Vessel d/p Instrumentation to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the Core Spray Sparger to Reactor Pressure Vessel d/p Instrumentation LCO and Surveillances may be relocated to a licensee controlled document.

APPENDIX A

2.1.A.1.c APRM ROD BLOCK TRIP SETTING
TABLE 3/4.2.C CONTROL ROD BLOCKS - APRM UPSCALE (FLOW BIASED, STARTUP
MODE), APRM DOWNSCALE

LSSS Statement:

The limiting safety system settings shall be as specified below:

A.1.c The APRM Rod Block Trip Setting shall be less than or equal to the limit specified in the COLR

LCO Statement:

The limiting conditions of operation for the instrumentation that initiates control rod blocks are given in Table 3.2.C.

Table 3.2.C Instrumentation that Initiates or Controls the Core and Containment Cooling Systems

3/4.2-25	APRM Upscale (Flow Biased)
3/4.2-25	APRM Upscale (Startup Mode)
3/4.2-25	APRM Downscale
3/4.2-25	APRM Inoperative

Discussion:

The Average Power Range Monitor (APRM) control rod blocks function to prevent a control rod withdrawal error during power range operations using LPRM signals to create the APRM rod block signal. APRMs provide information about the average core power and APRM rod blocks are not assumed to mitigate a DBA or transient.

Comparison to Screening Criteria:

1. The APRM control rod blocks are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. The APRM control rod block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. The APRM control rod blocks are not part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

APPENDIX A

2.1.A.1.c APRM ROD BLOCK TRIP SETTING
TABLE 3/4.2.C CONTROL ROD BLOCKS - APR. UPSCALE (FLOW BIASED, STARTUP
MODE), APRM DOWNSCALE (cont'd.)

4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 135) of NEDO 31466, and verified by TVA, the loss of the APRM control rod block functions was found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the APRM Control Rod Block Instrumentation LCO and Surveillances may be relocated to a licensee controlled document.

APPENDIX A

TABLE 3/4.2.C CONTROL ROD BLOCKS - IRM UPSCALE, IRM DOWNSCALE, IRM DETECTOR NOT IN STARTUP POSITION, IRM INOPERATIVE

LCO Statement:

The limiting conditions of operation for the instrumentation that initiate control rod blocks are given in Table 3.2.C.

Table 3.2.C Instrumentation that Initiates or Controls the Core and Containment Cooling Systems

3/4.2-25	IRM Upscale
3/4.2-25	IRM Downscale
3/4.2-25	IRM Detector Not In Startup Position
3/4.2-25	IRM Inoperative

Discussion:

The Intermediate Range Monitor (IRM) control rod blocks function to prevent a control rod withdrawal error during reactor startup using IRM signals to create the rod block signal. IRMs are provided to monitor the neutron flux levels during refueling and startup conditions. No design basis accident or transient analysis takes credit for rod block signals initiated by IRMs.

Comparison to Screening Criteria:

1. The IRM control rod blocks are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. The IRM control rod block instrumentation is not a process variable that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. The IRM control rod blocks are not part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 138) of NEDO-31466, and verified by TVA, the loss of the IRM control rod block functions was found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the IRM Control Rod Block Instrumentation LCO and Surveillances may be relocated to a licensee controlled document.

APPENDIX A

TABLE 3/4.2.C CONTROL ROD BLOCKS - SRM UPSCALE, SRM DOWNSCALE, SRM DETECTOR NOT IN STARTUP POSITION, SRM INOPERATIVE

LCO Statement:

The limiting conditions of operation for the instrumentation that initiates control rod blocks are given in Table 3.2.C.

Table 3.2.C Instrumentation that Initiates or Controls the Core and Containment Cooling Systems

3/4.2-25	SRM Upscale
3/4.2-25	SRM Downscale
3/4.2-25	SRM Detector Not In Startup Position
3/4.2-25	SRM Inoperative

Discussion:

The Source Range Monitor (SRM) control rod blocks function to prevent a control rod withdrawal error during reactor startup using SRM signals to create the rod block signal. SRM signals are used to monitor the neutron flux levels during refueling, shutdown, and startup conditions. No design basis accident or transient analysis takes credit for rod block signals initiated by the SRMs.

Comparison to Screening Criteria:

1. The SRM control rod blocks are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. The SRM control rod block instrumentation is not a process variable that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. The SRM control rod blocks are not part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 137) of NEDO-31466, and verified by TVA, the loss of the SRM control rod block functions was found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the SRM Control Rod Block Instrumentation LCO and Surveillances may be relocated to a licensee controlled document.

APPENDIX A

TABLE 3.2.C CONTROL ROD BLOCKS - SCRAM DISCHARGE INSTRUMENT VOLUME HIGH LEVEL

LCO Statement:

The limiting conditions for operation for the instrumentation that initiates control rod blocks are given in Table 3.2.C.

Table 3.2.C Instrumentation that Initiate Control Rod Blocks

3/4.2-25 Scram Discharge Instrument Volume High Level

Discussion:

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

Comparison to Screening Criteria:

1. The SDV control rod block is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. The SDV control rod block instrumentation is not a process variable that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. The SDV control rod block is not part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 139) of NEDO-31466, and verified by TVA, the loss of the Scram Discharge Volume High Level Control Rod Block Instrumentation was found to be a nonsignificant risk contributor to core damage frequency and offsite releases.

APPENDIX A

TABLE 3.2.C CONTROL ROD BLOCKS - SCRAM DISCHARGE INSTRUMENT VOLUME HIGH
LEVEL (cont'd.)

Conclusion:

Since the screening criteria have not been satisfied, the Scram Discharge Instrument Volume High Level Control Rod Block Instrumentation LCO and Surveillances may be relocated to a licensee controlled document.

APPENDIX A

3/4.2.F SURVEILLANCE INSTRUMENTATION

LCO Statement:

The limiting conditions for the instrumentation that provides surveillance information readouts are given in Table 3.2.F

Table 3.2.F Surveillance Instrumentation

3/4.2-31	Reactor Water Level
3/4.2-31	Reactor Pressure
3/4.2-31	Drywell Pressure
3/4.2-31	Drywell Air Temperature
3/4.2-31	Suppression Chamber Air Temperature
3/4.2-31	Control Rod Position
3/4.2-31	Neutron Monitoring
3/4.2-31	Drywell Pressure Alarm
3/4.2-31	Drywell Temperature and Pressure and Timer
3/4.2-31	CAD Tank Level
3/4.2-32	Drywell and Torus Hydrogen Concentration
3/4.2-32	Drywell to Suppression Chamber Differential Pressure
3/4.2-32	Relief Valve Tailpipe Thermocouple Temperature or Acoustic Monitor on Relief Valve Tailpipe
3/4.2-32	Primary Containment High Range Radiation Monitors
3/4.2-32	Drywell Pressure - Wide Range
3/4.2-32	Suppression Chamber Water Level - Wide Range
3/4.2-32	Suppression Pool Bulk Temperature
3/4.2-32	Wide Range Gaseous Effluent Radiation Monitor

Discussion:

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

Comparison to Screening Criteria:

The NRC position on application of the deterministic screening criteria to post-accident monitoring instrumentation is documented in letter dated May 7, 1988 from T. E. Murley (NRC) to R. F. Janecek (BWROG). The position taken was that the post-accident monitoring instrumentation table list should contain, on a plant specific basis, all Regulatory Guide 1.97 Type A instruments specified in the plants SER on Regulatory Guide 1.97, and all Regulatory Guide 1.97 Category 1 instruments. Accordingly, this position has been applied to the BFN Regulatory Guide 1.97 instruments. Those instruments not meeting this criteria have been relocated from the Technical Specifications to a licensee controlled document.

APPENDIX A

3/3.2.F SURVEILLANCE INSTRUMENTATION (cont'd.)

The following summarizes the BFN position for those instruments currently in Technical Specifications.

From NRC SER dated April 30, 1984, Subject: Conformance to RG 1.97

Category 1 or Type A Variables

1. Reactor Pressure
2. Reactor Vessel Water Level (wide range, accident range)
3. Suppression Pool Water Temperature
4. Suppression Pool Water Level (wide range)
5. Drywell Pressure (normal range, wide range)
6. Drywell Air Temperature
7. Primary Containment Area Radiation
8. Drywell and Torus Hydrogen Concentration

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

Conclusion:

Since the screening criteria have not been satisfied for instruments that do not meet Regulatory Guide 1.97 Type A variable requirements or Category 1 variable Type A instruments, their associated LCO and Surveillances will be relocated to a licensee controlled document. The instruments to be relocated are as follows:

1. Drywell Temperature and Pressure Timer
2. Suppression Chamber Air Temperature
3. Control Rod Position
4. Neutron Monitoring
5. Drywell Pressure Alarm
6. CAD Tank Level
7. Drywell to Suppression Chamber Differential Pressure
8. Relief Valve Tailpipe Thermocouple Temperature or Acoustic Monitor on Relief Valve Tailpipe
9. Wide Range Gaseous Effluent Radiation Monitor

APPENDIX A

3/4.6.B.1 - 5 PRIMARY SYSTEM BOUNDARY - COOLANT CHEMISTRY

LCO Statement:

The following limits shall be observed for reactor water quality prior to any startup and when operating at rated pressure:

- a) Conductivity at 25°C - 2.0 mho/cm
- b) Chloride concentration - 0.1 ppm

Discussion:

Poor reactor coolant water chemistry contributes to the long-term degradation of system materials and, thus, is not of immediate importance to the plant operator. Reactor coolant water chemistry is maintained to reduce the possibility of failure in the reactor coolant system pressure boundary caused by corrosion. In summary, the chemistry monitoring activity is of a long term preventive purpose rather than mitigative.

Comparison to Screening Criteria:

1. Reactor coolant water chemistry is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. Reactor coolant water chemistry is not a process variable that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. Reactor coolant water chemistry is not part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 211) of NEDO-31466, and verified by TVA, Coolant Chemistry requirements not being met was found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the Coolant Chemistry (Conductivity and Chloride) LCO and Surveillances may be relocated to a licensee controlled document.

APPENDIX A

3/4.6.G PRIMARY SYSTEM BOUNDARY - STRUCTURAL INTEGRITY

LCO Statement:

The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station.

Discussion:

The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of those components will be maintained throughout the components life. Operability of the primary system boundary is ensured by separate Technical Specifications and therefore, the inspections are not required to be retained in the Technical Specifications. This Technical Specification is more directed toward prevention of component degradation and continued long term maintenance of acceptable structural conditions. However, it is not necessary to retain this Specification to ensure the operability of the primary system boundary.

Comparison to Screening Criteria:

1. The inspections stipulated by this Specification are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. The inspections stipulated by this Specification do not monitor process variables that are initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. The ASME Code Class 1, 2, and 3 components inspected per this Specification are assumed to function to mitigate a DBA. Their capability to perform this function is addressed by other Technical Specifications. This Technical Specification, however, only specifies inspection requirements for these components; and these inspections can only be performed when the plant is shutdown. Therefore, Criterion 3 is not satisfied.
4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 216) of NEDO-31466, and verified by TVA, the lack of a Structural Integrity Specification was found to be a non-significant risk contributor to core damage frequency and offsite releases since the requirement is currently covered by 10 CFR 50.55a and the Inservice Inspection Program.

Conclusion:

Since the screening criteria have not been satisfied, the Structural Integrity LCO and Surveillance may be relocated to a licensee controlled document.

APPENDIX A

3/4.10.D REFUELING OPERATIONS - REACTOR BUILDING CRANE

LCO Statement

The reactor building crane shall be OPERABLE:

- a. When a spent fuel cask is handled.
- b. Whenever new or spent fuel is handled with the 5-ton hoist.

Discussion:

The reactor building crane and 125 ton hoist are required to be operable for handling of the spent fuel in the reactor building. This LCO specifies minimum operability requirements to prevent damage to the refueling platform equipment and core internals. The crane is not assumed to function to mitigate the consequences of a DBA.

Comparison to Screening Criteria:

1. The reactor building crane is not used, nor is it capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary (RCPB).
2. The reactor building crane is not a process variable that is an initial condition of a DBA or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. The reactor building crane is not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 287) of NEDO-31466, and verified by TVA, Reactor building crane requirements not being met was found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the LCO and associated surveillance may be relocated to the Technical Requirements Manual.

APPENDIX A

3/4.10.E REFUELING OPERATIONS - SPENT FUEL CRANE 3.10.F

LCO Statements

Spent Fuel Cask

Upon receipt, an empty fuel cask shall not be lifted until a visual inspection is made of the cask-lifting trunnions and fastening connection has been conducted.

Spent Fuel Cask Handling - Refueling Floor

Administrative control shall be exercised to limit the height the spent fuel cask is raised above the refueling floor by the reactor building crane to 6 inches, except for entry into the cask decontamination chamber where height above the floor will be approximately 3 feet.

The spent fuel cask yoke safety links shall be properly positioned at all times except when the cask is in the decontamination chamber.

Discussion:

BFN analysis has been performed to address the handling of spent fuel cask. However, BFN currently does not have the need to handle spent fuel cask. Therefore, these LCOs serve no useful purpose and should be deleted.

Comparison to Screening Criteria:

1. The Spent fuel cask and spent fuel cask handling controls are not used to detect, and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary (RCPB).
2. The Spent fuel cask and spent fuel cask handling controls are not capable of monitoring a process variable that is an initial condition of a DBA or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. The Spent fuel cask and spent fuel cask handling controls are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 287) of NEDO-31466, and verified by TVA, Spent fuel cask and spent fuel cask handling control requirements not being met was found to be a non-significant risk contributor to core damage frequency and offsite releases.

APPENDIX A

3/4.10.E REFUELING OPERATIONS - SPENT FUEL CRANE
3.10.F (cont'd.)

Conclusion:

Since the screening criteria have not been satisfied and the LCOs serve no useful purpose, the LCOs and associated surveillance may be deleted.

APPENDIX A

3/4.2.H

FLOOD PROTECTION INSTRUMENTATION

LCO Statement:

The unit shall be shutdown and placed in the cold condition when Wheeler Reservoir lake stage rises to a level such that water from the reservoir begins to run across the pumping station deck at elevation 565. Requirements for the instrumentation that monitors the reservoir level are given in Table 3.2.H.

Discussion:

Provides capability to predict flood levels of large magnitudes which allows the plant to take advantage of advance warning to take appropriate action when reservoir levels above normal pool are predicted.

Comparison to Screening Criteria:

1. The Reservoir Level Monitoring instrumentation is not used to detect, and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary (RCPB).
2. The Reservoir Level Monitoring instrumentation is not capable of monitoring a process variable that is an initial condition of a DBA or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. The Reservoir Level Monitoring instrumentation is not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 273) of NEDO-31466, and verified by TVA, Reservoir Level Monitoring instrumentation requirements not being met was found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the LCO and associated surveillance may be relocated to a licensee controlled document.

APPENDIX A

3/4.8.E

MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

LCO Statement:

The leakage test shall be capable of detecting presence of 0.005 microcurie of radioactive material on the test sample.

Discussion:

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

Comparison to Screening Criteria:

1. Miscellaneous radioactive materials sources requirements are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. Miscellaneous radioactive materials sources requirements are not process variables that are initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. Miscellaneous radioactive materials sources requirements are not part of the primary success path that function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 267) of NEDO 31466, and verified by TVA, the Miscellaneous Radioactive Materials Sources requirements not being met was found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the Miscellaneous Radioactive Materials Sources LCO and Surveillances may be relocated to a licensee controlled document.

APPENDIX A

3/4.2.J

SEISMIC MONITORING INSTRUMENTATION

LCO Statement:

The seismic monitoring instrumentation shown in Table 3.2.J shall be operable.

Discussion:

In the event of an earthquake, seismic monitoring instrumentation is required to determine the magnitude of the seismic event. These instruments do not perform any automatic action. They are used to measure the magnitude of the seismic event for comparison to the design basis of the plant to ensure the design margins for plant equipment and structures have not been violated. Since the determination of the magnitude of the seismic event is performed after the event has occurred, this instrumentation has no bearing on the mitigation of any design basis accident (DBA) or transient.

Comparison to Screening Criteria:

1. Seismic monitoring instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. Seismic monitoring instrumentation is not a process variable that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. Seismic monitoring instrumentation is not part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 151) of NEDO-31466, and verified by TVA, the loss of the Seismic Monitoring instrumentation was found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the Seismic Monitoring Instrumentation LCO and Surveillances may be relocated to a licensee controlled document.

APPENDIX A

3.2.F/4.2.F METEOROLOGICAL MONITORING INSTRUMENTATION

LCO Statement

The meteorological monitoring instrumentation listed in Table 3.2.I shall be OPERABLE at all times.

Discussion:

Ensures that there is a sufficient amount of data available to estimate potential radiological doses. There are no automatic actions during any event that these instruments perform, nor do they actuate to mitigate a DBA or transient.

Comparison to Screening Criteria:

1. The Meteorological Monitoring instrumentation is not used to detect, and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary (RCPB).
2. The Meteorological Monitoring instrumentation is not capable of monitoring a process variable that is an initial condition of a DBA or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. The Meteorological Monitoring instrumentation is not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 151) of NEDO-31466 (Table 4-1 and 6-3, Item 152), the loss of meteorological monitoring instrumentation is a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the LCO and associated surveillance may be relocated to the Technical Requirements Manual.

APPENDIX A

3.8.B.9, 10
4.8.B.5

RADIOACTIVE MATERIALS - AIRBORNE EFFLUENTS, EXPLOSIVE GAS MIXTURE

LCO Statement:

The concentration of hydrogen downstream of the recombiners shall be limited to less than or equal to 4% by volume.

Discussion:

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

Comparison to Screening Criteria:

1. The explosive gas mixture requirements are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. The explosive gas mixture requirements are not process variables that are initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. The explosive gas mixture requirements are not part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 306) of NEDO-31466, and verified by TVA, an explosive gas mixture in the waste gas holdup system was found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the Explosive Gas Mixture LCO and Surveillances may be relocated to a licensee controlled document.

APPENDIX A

3.2.F/4.2.F EXPLOSIVE GAS MONITORING INSTRUMENTATION

LCO Statement

The explosive gas monitoring instrumentation listed in Table 3.2.K shall be OPERABLE with the applicability as shown in Tables 3.2.K/4.2.K.

Discussion:

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

Comparison to Screening Criteria:

1. The explosive gas monitoring instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. The explosive gas monitoring instrumentation is not used to monitor a process variables that is an initial conditions of a DBA or transient. Excessive system effluent is not an indication of a DBA or transient.
3. The explosive gas monitoring instrumentation is not part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (Item 306) of NEDO-31466, and verified by TVA, an explosive gas mixture in the waste gas holdup system was found to be a non-significant risk contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the Explosive Gas Mixture LCO and Surveillances may be relocated to a licensee controlled document.

APPENDIX A

3.10.A.3 & 4 REFUELING PLATFORM EQUIPMENT INTERLOCKS

LCO Statement

Refueling Interlocks

3. The fuel grapple hoist load switch shall be set at $\leq 1,000$ lbs.
4. If the frame-mounted auxiliary hoist, the monorail-mounted auxiliary hoist, or the service platform hoist is to be used for handling fuel with the head off the reactor vessel, the load limit switch on the hoist to be used shall be set at < 400 lbs.

Discussion:

Specifies minimum operability requirements. Designed to provide the capabilities to prevent damage to the refueling platform equipment and core internals, they are not assumed to function to mitigate the consequences of a DBA.

Comparison to Screening Criteria:

1. Refueling platform equipment interlocks are not used to detect, and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary (RCPB).
2. Refueling platform equipment interlocks are not capable of monitoring a process variable that is an initial condition of a DBA or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. Refueling platform equipment interlocks are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. As discussed in Sections 3.5 and 6 and summarized in Table 4-1 (item 306) of NEDO-31466, and verified by TVA, the loss of the refueling platform equipment interlocks is a non-significant contributor to core damage frequency and offsite release.

Conclusion:

Since the screening criteria have not been satisfied, the LCO and associated surveillance may be relocated to a licensee controlled document..

APPENDIX B
BFN SPECIFIC
RISK SIGNIFICANT EVALUATIONS

APPENDIX B

TABLE 3/4.2.B CORE SPRAY SPARGER TO REACTOR PRESSURE VESSEL d/p

LCO Statement:

The limiting conditions for operation for the instrumentation that initiates or controls the core and containment cooling systems are given in Table 3.2.B.

Table 3.2.B Instrumentation that Initiates or Controls the Core and Containment Cooling Systems

3/4.2-17 Core Spray Sparger to Reactor Pressure Vessel d/p

Description of Requirement:

This instrumentation measures the differential pressure between the core spray sparger and the reactor pressure vessel above the core plate and alarms if a break is detected. This instrumentation does not actuate any equipment.

Risk Justification:

The function of the instrumentation is to identify a break in the core spray sparger. The probability of a pipe break (EPRI TR-100380) is extremely low, therefore the relative probability as defined in NEDO-31466 is low. LOCAs represent a small contribution to the BFN core damage frequency (CDF). A break in the sparger in the reactor pressure vessel would, without a LOCA, provide injection to the core. Given the success of injection with a break in non-LOCA accidents and the small contribution of LOCA to the CDF, the relative significance from an offsite radiological dose perspective would be low. The risk category would therefore be considered non-significant (NS).

<u>Relative Probability</u>	<u>Relative Significance</u>	<u>Risk Category</u>
Low	Low	NS

APPENDIX B

3/4.2. FLOOD PROTECTION

LCO Statement

The unit shall be shutdown and placed in the cold condition when Wheeler Reservoir lake stage rises to a level such that water from the reservoir begins to run across the pumping station deck at elevation 565. Requirements for the instrumentation that monitors the reservoir level are given in Table 3.2.H.

Description of Requirements:

This Technical Specification has provisions for high reservoir water level instrumentation. A high reservoir water level indication is a preliminary indication of a flood. A flood is not a design basis accident or transient, thus reservoir water level is not credited in the safety analysis.

Risk Justification:

An analysis of the risk of external flooding was performed in BFN's Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities. Historical flood data was collected and analyzed to determine the frequency and magnitude of floods at BFN. All critical equipment essential to the safe shutdown of the plant are flood protected to an elevation well above that required in the current LCO. Given the plant design features and the conservative analysis of flooding in the IPEEE, the contribution of flooding to overall plant risk (probability of occurrence and radiological consequence) is considered negligible.

<u>Relative Probability</u>	<u>Relative Significance</u>	<u>Risk Category</u>
Low	Low	NS

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BROWNS FERRY NUCLEAR PLANT

IMPROVED STANDARD TECHNICAL SPECIFICATIONS

Enclosure V
Volume 18

TENNESSEE VALLEY AUTHORITY

3.6/4.6 PRIMARY SYSTEM BOUNDARY

NOV 18 1988

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

See Justification for Changes for BFN ISTS 3.4.2 and 3.4.3

3.6.E. Jet Pumps

- 1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.D. Relief Valves

- 3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.
- 4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

- 1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

SR 3.4.1.1

(A1)

Proposed Note for SR 3.4.1.1
 Verify ~~The two recirculation loops have a flow 10% imbalance of 15% or less more when the pumps are operated at the same speed.~~
 mismatch
 jet pump
 (M2)

both recirculation loops

- b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.
- c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

AUG 04 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

See Justification for Changes to BFN 1STS 3.4.2

(L1)

Proposed Note to SR 3.4.1.1

4.6.E. Jet Pumps

2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

~~3.6.F Recirculation Pump Operation~~
LCO 3.4.1 ~~matched flow requirement~~

(A1)

ACTIONS C+D

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.

(LA1)

2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

ACTION D

3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation

See Justification for Changes for BFN 1STS 3.4.9

~~4.6.F Recirculation Pump Operation~~

SR 3.4.1.1

(M2)

1. Recirculation pump speeds shall be checked and logged at least once per day.

(A3)

(LA2)

(A1)

~~2. No additional surveillance required.~~

(LA2)

3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.

MAY 31 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.F Recirculation Pump Operation~~

~~3.6.F.3 (Cont'd)~~

temperature of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

ACTION D

see Justification for Changes for BFN ISTS 3.4.9

(m1)

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

ACTION E

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.

a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

See Justification for Changes for CTS 3.6.G/4.6.G in this Section



MAY 31 1994

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.M Core Thermal Hydraulic Stability~~

~~4.5.M Core Thermal Hydraulic Stability~~

LCO
3.4.1

1. The reactor shall not be operated at a thermal power and core flow inside of Regions I and II of Figure 3.5.M-1.

Action
A

2. If Region I of Figure 3.5.M-1 is entered, immediately initiate a manual scram.

Action
B

3. If Region II of Figure 3.5.M-1 is entered:

a. Immediately initiate action and exit the region within 2 hours by inserting control rods or by increasing core flow (starting a recirculation pump to exit the region is not an appropriate action), and

LA3

b. While exiting the region, immediately initiate a manual scram if thermal-hydraulic instability is observed, as evidenced by APRM oscillations which exceed 10 percent peak-to-peak of rated or LPRM oscillations which exceed 30 percent peak-to-peak of scale. If periodic LPRM upscale or downscale alarms occur, immediately check the APRM's and individual LPRM's for evidence of thermal-hydraulic instability.

SR 3.4.1.2

1. Verify that the reactor is outside of Region I and II of Figure 3.5.M-1:

a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and

b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

Proposed Action D

L2

LA3

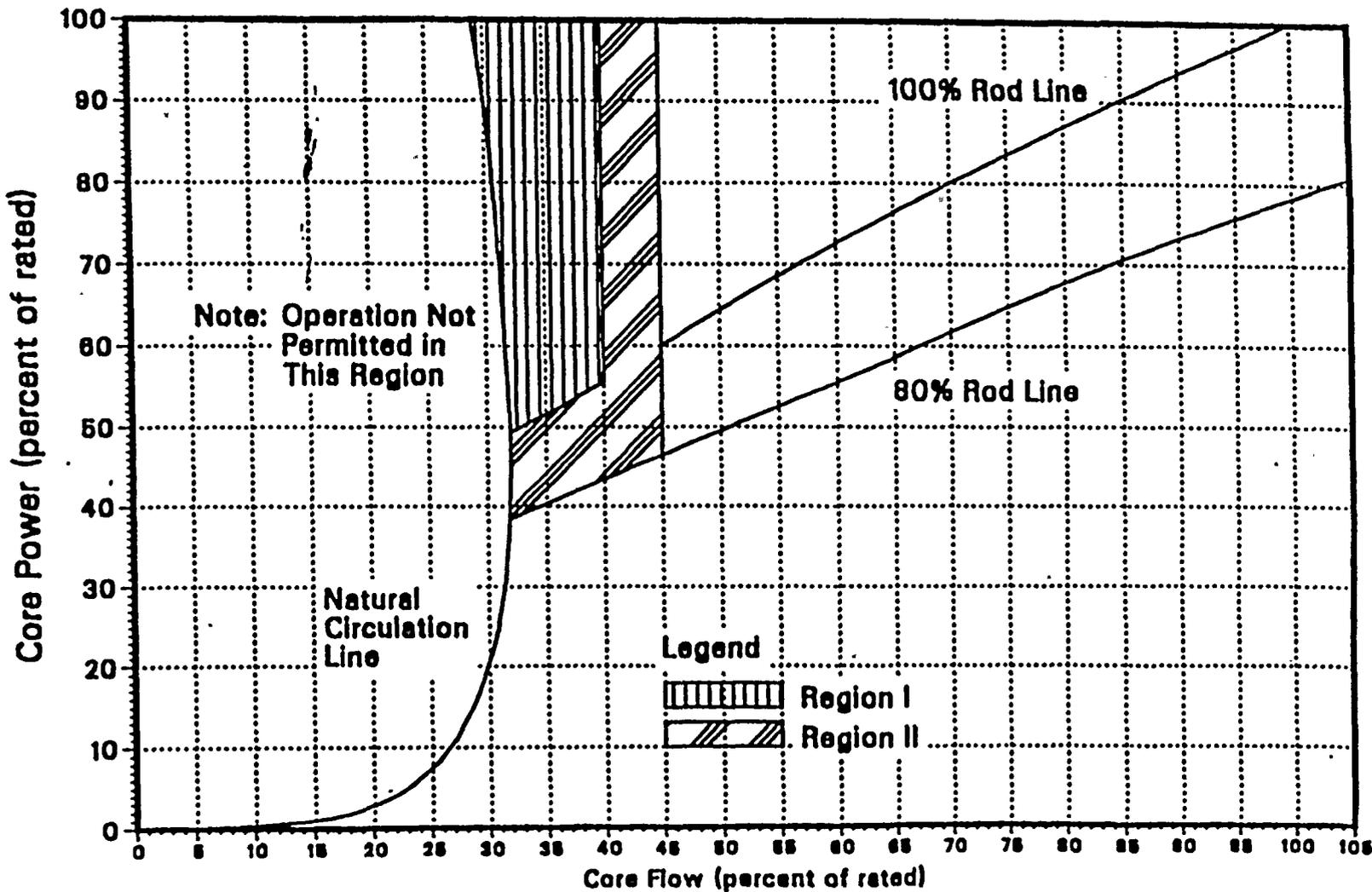


A1

3.4.1-1

Figure ~~3.5.M-1~~

BFN Power/Flow Stability Regions



BFN
Unit 1

3.5.14.5.22

AMENDMENT NO. 208

Specification 3.4.1

MAY 31 1994

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

See Justification
for Changes for
BFN ISTS 3.4.2
and 3.4.3

3.6.E. Jet Pumps

1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be shutdown in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.D. Relief Valves

3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.
4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

L1
Proposed Note
for SR 3.4.1.1

SR 3.4.1.1

A1

- a. ~~The two recirculation loops have a flow mismatch is imbalance of 15% or less when the pumps are operated at the same speed.~~ ^{Verify jet pump} ^{100%} ^{M2} ^{both recirculation loops}
- b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.
- c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.



AUG 04 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

See Justification for Changes to BFN ISTS 3.4.2

4.6.E. Jet Pumps

2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

(L1)
Proposed Note to SR 3.4.1.1

~~3.6.F Recirculation Pump Operation~~

LCO 3.4.1 ~~matched flow requirement~~ (M2)

ACTIONS C+D

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.

(LAI) 2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. (M1)

ACTION: D During such interval, restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature of the reactor

~~4.6.F. Recirculation Pump Operation~~

SR 3.4.1.1

(A2)

1. (M2) Recirculation pump speeds shall be checked and logged at least once per day.

(A3) (LA2)

(A1) 2. ~~No additional surveillance required.~~

(LA2) 3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.

See Justification for Changes for BFN ISTS 3.4.9



MAR 18 1993

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(AI)

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.F Recirculation Pump Operation~~

~~3.6.F.3 (Cont'd)~~

vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

ACTION D

(MI)

(See Justification for Changes for BFN 1STS 3.4.9)

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

ACTION E

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a COLD SHUTDOWN CONDITION or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 50.55, except where specific written relief has been granted by 10 CFR 50, Section 50.55a(g)(6)(i).
2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

(See Justification for Changes for CTS 3.6.G/4.6.G in this Section)

3.6/4.6-13



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 Core and Containment Cooling Systems~~

~~4.5 Core and Containment Cooling Systems~~

L. APRM Setpoints

1. Whenever the core thermal power is $\geq 25\%$ of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0 , or the APRM scram setpoint equation listed in Section 2.1.A and the APRM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT shall be multiplied by FRP/CMFLPD.
2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.
3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to $\leq 25\%$ of rated thermal power within 4 hours.

L. APRM Setpoints

FRP/CMFLPD shall be determined daily when the reactor is $\geq 25\%$ of rated thermal power.

See Justification for Changes for BFN ISTS 3.2.4

~~M. Core Thermal Hydraulic Stability~~

LCO
3.4.1

1. The reactor shall not be operated at a thermal power and core flow inside of Regions I and II of Figure 3.5.M-1.
2. If Region I of Figure 3.5.M-1 is entered, immediately initiate a manual scram.
3. If Region II of Figure 3.5.M-1 is entered:

ACTION
A

ACTION
B

(A1)

~~M. Core Thermal Hydraulic Stability~~
SR 3.4.1.2

1. Verify that the reactor is outside of Region I and II of Figure 3.5.M-1:
 - a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and
 - b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

OCT 05 1989

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 Core and Containment Cooling Systems~~

~~4.5 Core and Containment Cooling Systems~~

3.5.M.3. (CONT'D)

ACTION B

- a. Immediately initiate action and exit the region within 2 hours by inserting control rods or by increasing core flow (starting a recirculation pump to exit the region is not an appropriate action), and
- b. While exiting the region, immediately initiate a manual scram if thermal-hydraulic instability is observed, as evidenced by APRM oscillations which exceed 10 percent peak-to-peak of rated or LPRM oscillations which exceed 30 percent peak-to-peak of scale. If periodic LPRM upscale or downscale alarms occur, immediately check the APRM's and individual LPRM's for evidence of thermal-hydraulic instability.

Proposed ACTION D

(L2)

LA3

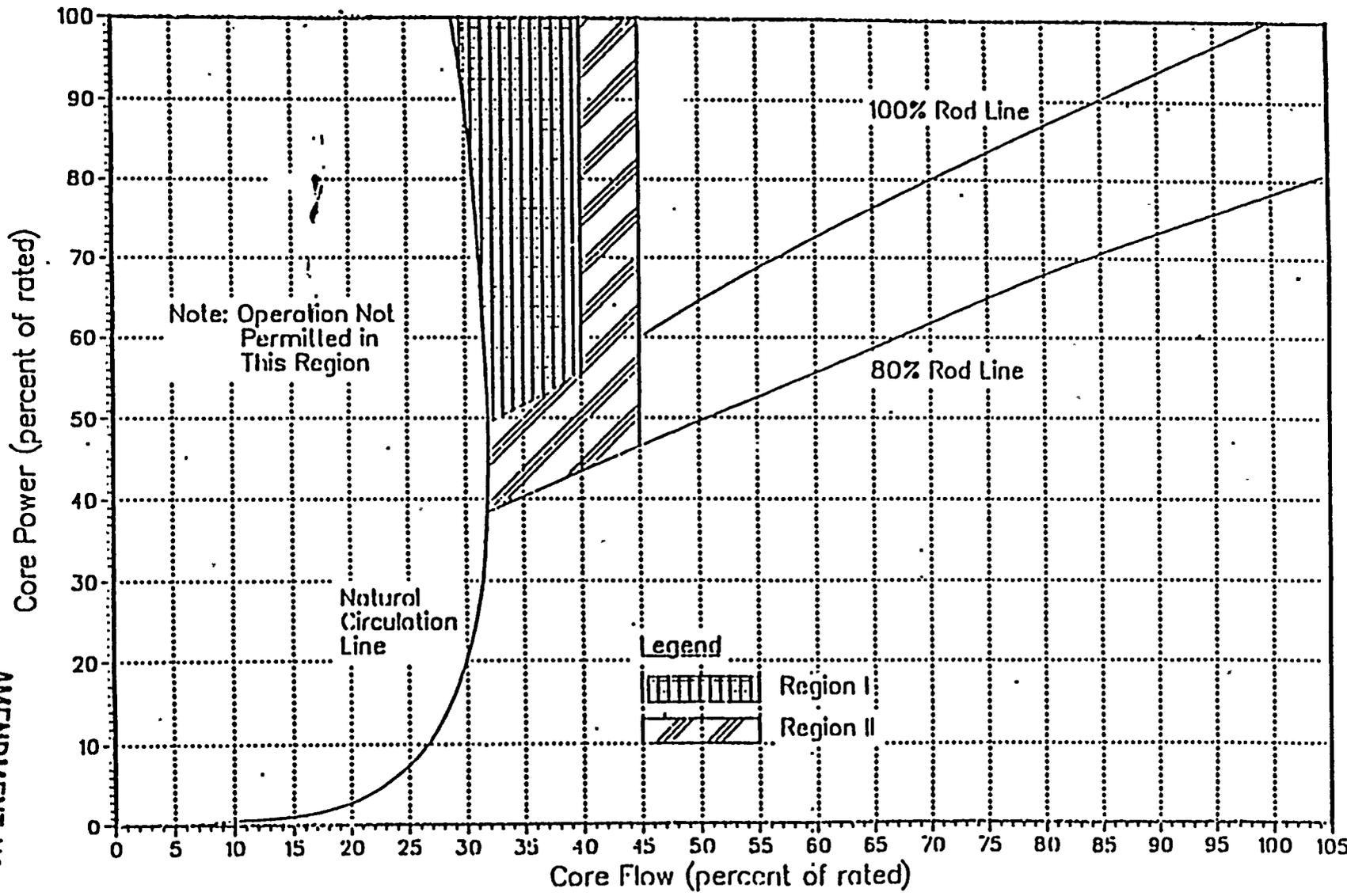
LA3



AI

3.4.1-1

Figure 3.5.M-P
BFN Power/Flow Stability Regions



BFN
Unit 2

3.5/L.S-2.2.8

AMENDMENT NO. 17.4

Specification 3.4.1
OCT 05 1989



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

NOV 18 1988

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

See Justification for changes for BFN ISTS 3.4.2 and 3.4.3

3.6.E. Jet Pumps

- 1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is INOPERABLE, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.D. Relief Valves

- 3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.
- 4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

- 1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

M2

SR3.4.1.1

Verify jet pump

Proposed Note for SR3.4.1.1

is 10% mismatch

or 15% or more when the pumps are in operation at the same speed.

L1

both recirculation loops

- a. The two recirculation loops have a flow mismatch of 15% or more when the pumps are in operation at the same speed.

- b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

- c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.



AUG 04 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

See Justification for Changes to BFN ISTS 3.4.2

(L1)

Proposed Note to SR 3.4.1.1

4.6.E. Jet Pumps

2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

(A1)

~~3.6.F Recirculation Pump Operation~~

LCO 3.4.1 (Matched Flow requirements) (M2)

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.

ACTIONS C+D

(A2)

2. Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

(LA1)

3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature

ACTION D

(M1)

See Justification for Changes for BFN ISTS 3.4.9 3.6/4.6-12

~~4.6.F Recirculation Pump Operation~~

SR 3.4.1.1

1. Recirculation pump speeds shall be checked and logged at least once per day.

(M2)

(A3)

(LA2)

(A1)

~~2. No additional surveillance required.~~

(LA2)

3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.F Recirculation Pump Operation~~

~~3.6.F.3 (Cont'd)~~

of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

ACTION D

See Justification for changes for BFN ISTS 3.4.9

(m1)

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

ACTION E

3.6.G Structural Integrity

- 1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

4.6.G Structural Integrity

- 1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
- 2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

See Justification for changes for CTS 3.6.G/4.6.G in this section



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

MAY 31 1994

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

~~3.5.M Core Thermal-Hydraulic Stability~~

~~4.5.M Core Thermal Hydraulic Stability~~

SR 3.4.1.2

LCO
3.4.1

Action
A

Action
B

1. The reactor shall not be operated at a thermal power and core flow inside of Regions I and II of Figure 3.5.M-1.

2. If Region I of Figure 3.5.M-1 is entered, immediately initiate a manual scram.

3. If Region II of Figure 3.5.M-1 is entered:

a. Immediately initiate action and exit the region within 2 hours by inserting control rods or by increasing core flow (starting a recirculation pump to exit the region is not an appropriate action), and

b. While exiting the region, immediately initiate a manual scram if thermal-hydraulic instability is observed, as evidenced by APRM oscillations which exceed 10 percent peak-to-peak of rated or LPRM oscillations which exceed 30 percent peak-to-peak of scale. If periodic LPRM upscale or downscale alarms occur, immediately check the APRM's and individual LPRM's for evidence of thermal-hydraulic instability.

1. Verify that the reactor is outside of Region I and II of Figure 3.5.M-1:

a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and

b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

(L2)

Proposed Action D

LA3

LA3

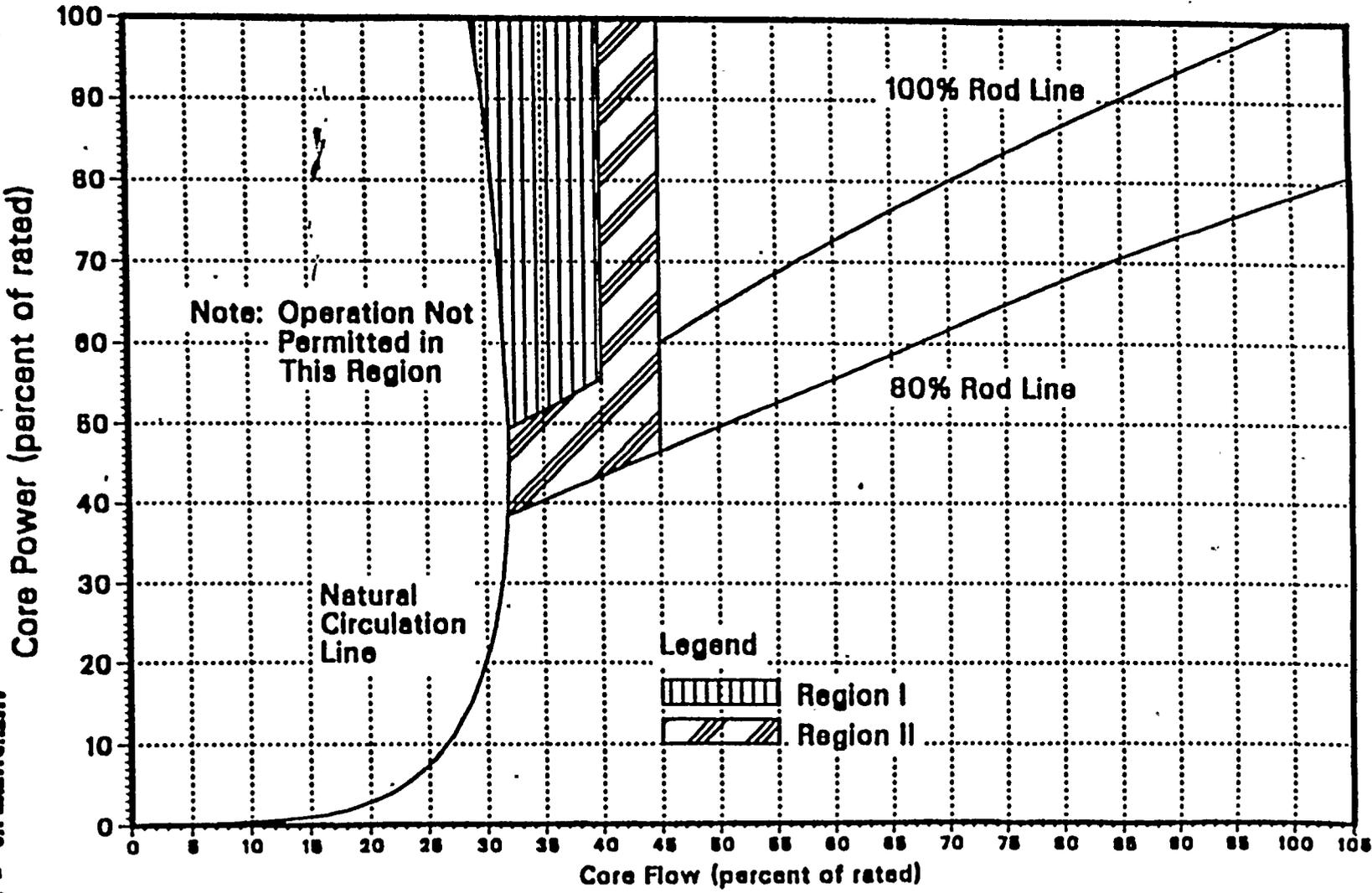


(A1)

3.4.1-1

~~Figure 3.5.M-1~~

BFN Power/Flow Stability Regions



BFN
Unit 3

3.5/4.5-21

AMENDMENT NO. 179

Specification 3.4.1

MAY 31 1994



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.1 - RECIRCULATION LOOPS OPERATING**

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 CTS requires the plant to be placed in the HOT SHUTDOWN CONDITION in 24 hours with one recirculation loop out of service. Proposed ACTION C requires the loop be returned to service in 12 hours or ACTION D requires the plant to be in MODE 3 (Hot Shutdown) in 12 hours. The CTS and the proposed ISTS Completion Times are essentially equivalent since both require the plant to be in MODE 3 in 24 hours.
- A3 The frequency for this Surveillance has been changed from once per day to once per 24 hours. This is a terminology change and is therefore administrative.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 CTS allows up to 24 hours operation with the reactor power < 1% with no recirculation loops operating (the total elapsed time in natural - circulation and one pump operation must be no greater than 24 hours). Proposed ACTION D is more restrictive since the time limit of 12 hours applies to < 1% while in MODE 2 also.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.1 - RECIRCULATION LOOPS OPERATING

- M2 The flow imbalance limit is being reduced to 10% of rated core flow when operating at < 70% of rated core flow, and to 5% of rated core flow when operating at \geq 70% of rated core flow. The current requirement is 15% mismatch of flow at the given flow conditions. While the limit appears to be less restrictive if core flow is \leq 66% of rated core flow, it is more restrictive when $>$ 66% of rated core flow (i.e., 15% x 66% or less is \leq 10% of rated core flow), where the unit normally operates. In addition, currently, this is only a problem if there is an imbalance in combination with two other conditions (CTS 4.6.B.1.b and c). The new requirement is separate from the other two, thus, actions will now be required if there is an imbalance by itself. Therefore, this change is considered more restrictive on plant operations.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

- LA1 This requirement is being relocated to plant specific procedures. The purpose of this limitation is to provide assurance that when shifting from one to two loop operations, excessive vibration of the jet pump risers will not occur. Short term excessive vibration should not result in immediate inoperability of a jet pump, but could reduce the lifetime of the jet pump. This type of requirement is generally found in plant operating procedures, similar to other operating requirements necessary to minimize the potential of damage to components. Changes to the procedures will be controlled by the licensee controlled programs.
- LA2 This requirement is being relocated to plant specific procedures. Details of the methods for performing this Surveillance, and any requirement to record data, has been relocated to plant procedures. Any changes to the procedures will be controlled by the licensee controlled programs.
- LA3 These requirements are being relocated to plant specific procedures. The details of the acceptable method for meeting an action requirement and what constitutes evidence of thermal hydraulic instability and the need to check for it have been relocated to plant procedures. Any changes to the procedures will be controlled by the licensee controlled programs.



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.1 - RECIRCULATION LOOPS OPERATING**

"Specific"

- L1 This change adds a note which states the Surveillance is not required to be performed until 24 hours after both recirculation loops are in operation. The Surveillance is not required to be performed until both loops are in operation since the mismatch limits are meaningless during single loop or natural circulation operation. Also, the Surveillance is allowed to be delayed 24 hours after both recirculation loops are in operation. This allows time to establish appropriate conditions for the test to be performed.
- L2 Per CTS 3.5.M.3.a, if Region II of Figure 3.5.M-1 is not exited within 2 hours, the Specification is violated and CTS 1.0.C.1 applies requiring the plant be placed in Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours. This provides actions for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. The BFN ISTS provides Action within the Specification which could be considered less restrictive than CTS. Action D allows 12 hours to be in MODE 3 (Hot Shutdown) and 36 hours to be in MODE 4 (Cold Shutdown). The proposed Action is considered less restrictive since 12 hours is allowed to place the unit in Hot Shutdown versus the 6 hours allowed to place the unit in Hot Standby per CTS.

UNIT 1

CURRENT
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NOV 18 1988

~~3.6/A.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

See justification for changes for BFN ISTS 3.4.3

4.6.D. Relief Valves

- The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.
- At least one relief valve shall be disassembled and inspected each operating cycle.

(A1)

~~3.6.E Jet Pumps~~
LCO 3.4.2

Applicability: Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.



See justification for changes for BFN ISTS 3.4.1

(L2) Proposed SR 3.4.2.1 Notes

(M2) Proposed SR 3.4.2.1 a & c

(A1) ~~E. Jet Pumps~~ Verify at least one of the following criteria is satisfied for each operating recirculation loop.

(A2) Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

(M2) b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

SR 3.4.2.1 b, d. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

(A3) the established pattern

(L3) less (A2)



AUG 04 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.6.E Jet Pumps~~

(A4)

SR 3.4.2.1

b. 2.

Applicability

Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than $\frac{10\%}{30}$.

(A5)

(A2)

(A3)

(L3)

less (A2)

See Justification for Changes for BFN ISTS 3.4.1

3.6.F Recirculation Pump Operation

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.
2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.
3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation

4.6.F Recirculation Pump Operation

1. Recirculation pump speeds shall be checked and logged at least once per day.
2. No additional surveillance required.
3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.



UNIT 2

CURRENT
TECHNICAL
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LIMITING CONDITIONS FOR OPERATION

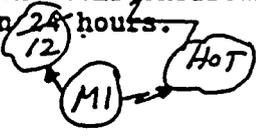
SURVEILLANCE REQUIREMENTS

(A1)

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.4.3

(A1)
3.6.E. Jet Pumps
LCO 3.4.2
+1
Applicability

Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours an orderly shutdown shall be initiated and the reactor shall be shutdown in the GOLD SHUTDOWN CONDITION within 24 hours.



SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.4.1

(L2) Proposed SR 3.4.2.1 Notes

(M2) Proposed SR 3.4.2.1 a + c

4.6.D. Relief Valves

- 3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.
- 4. At least one relief valve shall be disassembled and inspected each operating cycle.

(A1) E. Jet Pumps
Verify at least one of the following criteria is satisfied for each operating recirculation loop

Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

(A2)

a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

(M2)

SR 3.4.2.1 b/c

(A3) The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

the established pattern

(A2) less

(L3) 20

AUG 04 1994

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

~~4.6.E. Jet Pumps~~

SR 3.4.2.1

b2.

Applicability

Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

(A4)

(A5)

(A2)

(A3)

(20)

(L3)

Flies

(AZ)

See Justification for Changes for BFN ISTS 3.4.1

3.6.F Recirculation Pump Operation

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.
2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.
3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval, restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature of the reactor

4.6.F. Recirculation Pump Operation

1. Recirculation pump speeds shall be checked and logged at least once per day.
2. No additional surveillance required.
3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



NOV 18 1988

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

See Justification for changes for BFN ISTS 3.4.3

4.6.D. Relief Valves

- The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.
- At least one relief valve shall be disassembled and inspected each operating cycle.

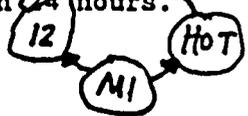
(A1)

~~3.6.E. Jet Pumps~~

LC0 3.4.2

Applicability

Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is INOPERABLE, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.



(L1)

See Justification for changes For BFN ISTS 3.4.1

(L2) Proposed SR 3.4.2.1 Notes

(M2) Proposed SR 3.4.2.1 a + c

(A1)

~~E. Jet Pumps~~

Verify at least one of the following criteria is satisfied for each operating recirculation loop

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

(A2)

a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

(A2)

b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

(M2)

SR 3.4.2.1

b. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than ~~±0% - 20%~~ less

(A3)

the established pattern

(A1)

~~4.6.2. Jet Pumps~~
~~SR 3.4.2.1~~

(A4)

b-2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than ~~10%~~ 20% (L3) (L55) (A2)

Applicability →

(A5)

(A3)

See Justification for changes for BFN 15 TS 3.4.1

3.6.F Recirculation Pump Operation

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.
2. Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.
3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature

4.6.F Recirculation Pump Operation

1. Recirculation pump speeds shall be checked and logged at least once per day.
2. No additional surveillance required.
3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.2 - JET PUMPS**

ADMINISTRATIVE

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change..

- A2 The wording of the surveillance was changed to require verification that one of the following criteria are met rather than verifying that none of the conditions exist simultaneously. This is consistent with NUREG-1433 which attempts to phrase everything in a positive manner. Due to the change in phrasing of the Surveillance, "more than" was changed to "less than or equal to" in criteria b and c.
- A3 The variance of the diffuser-to-lower plenum differential pressure reading on an individual jet pump will now be taken from the established pattern rather than from the mean of all jet pump differential pressures. This change is in accordance with the recommendations of SIL-330 and NUREG/CR-3052 and is consistent with NUREG-1433.
- A4 The conditions of the Surveillance Requirement are assured by LCO 3.4.1. Therefore, there is no need to restate the conditions for jet pump operability.
- A5 The frequency for this Surveillance has been changed from daily to once per 24 hours. This is a terminology change and is therefore administrative.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.2 - JET PUMPS

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 The requirement to place the plant in a Cold Shutdown condition within 24 hours when a jet pump is inoperable has been revised to reflect placing the plant in a non-applicable condition. Current Specification 1.0.C.1 states action requirements are applicable during the operational conditions of each specification. Therefore, the requirement to place the plant in Cold Shutdown is not applicable after Mode 3 is reached. The revised action requires plant power to be brought to Mode 3 (outside the applicable condition) within 12 hours. The current action allows 24 hours to place the plant in a non-applicable condition. As such, this is an additional restriction on plant operation which constitutes a more restrictive change.
- M2 This change adds two requirements to the Surveillance to detect significant degradation in jet pump performance that precedes jet pump failure. The first requirement added would detect a change in the relationship between pump speed, and pump flow and loop flow (difference > 5%). A change in the relationship indicates a plug flow restriction, loss in pump hydraulic performance, leakage, or new flow path between the recirculation pump discharge and jet pump nozzle. The second requirement added monitors the jet pump flow versus established patterns. Any deviations > 10% from normal are considered indicative of potential problem in the recirculation drive flow or jet pump system. These two added requirements to the Surveillance help to detect significant degradation in jet pump performance that precedes jet pump failure. Requirements added to Surveillance Requirements constitute a more restrictive change. In addition, CTS 4.6.E.1 allows jet pump operability to be verified by demonstrating that the two recirculation loops have a flow imbalance of $\leq 15\%$ when the pumps are operated at the same speed. This is now a separate requirement (Proposed SR 3.4.1.1 - See M2 of the Justification for Changes for Specification 3.4.1) and can no longer be used by itself to demonstrate jet pump operability. This change is consistent with NUREG-1433.

SIL-330 provides two alternate testing criteria (thus the deletion of current Surveillance 4.6.E.1.b). One method uses easy to perform surveillances with strict limits to initially screen jet pump operability (the proposed changes above). If these limits are not met, another set of Surveillances exist (current Technical Specifications). Revising the Surveillances to separate the flow imbalance test requirement and to include the stricter limits reflects a more restrictive change.



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.2 - JET PUMPS**

TECHNICAL CHANGE - LESS RESTRICTIVE

"Specific"

L1 This change deletes the current shutdown requirement associated with jet pump flow indication. Currently, when required jet pump flow indication is lost, an orderly shutdown must be initiated in 12 hours and the reactor is required to be in Cold Shutdown within the following 24 hours (since Mode 3 is the non-applicable mode, then 24 hours is allowed to reach Mode 3; see discussion of change M1 for ITS 3.4.2). The proposed Specification implicitly requires the jet pump flow indication to be operable only for the performance of the Surveillance Requirement. If the flow indication is inoperable when the surveillance is required to be performed and jet pump flow can not be determined by other means, the jet pump would be declared inoperable and the appropriate actions would be followed. Since the proposed jet pump surveillance requirement is required to be performed every 24 hours (the 25% extension per SR 3.0.2 can be applied) and the Required Actions require the reactor to be in Mode 3 within 12 hours, the maximum difference in the current Specification and the proposed specification is 6 hours. As a result, the proposed specification effectively allows a maximum of an additional 6 hours (which is the 25% extension) to reach a non-applicable Mode if a required core flow indicator is inoperable and jet pump flow can not be determined. Depending on when the failure occurs, 6 hours is the maximum increase over the current Specifications (failure occurring immediately after the surveillance is performed). The following table provides the details of the calculation of the 6 hour period:

Current Tech Specs	Proposed Tech Specs
Time 0 hours- Jet Pump Indication Fails - 12 hr AOT Begins	Time 0 hours - Jet Pump Indication Fails (Immediately After SR
Time 12 hours- 12 hr AOT Expires - 24 hr AOT Begins to MODE 3 (per 3.0.A; see M1)	Time 30 hours- SR due; Flow Indication Inop (24 hrs x 1.25) - 12 hr AOT to MODE 3 Begins
Time 36 hours- 24 hr AOT Expires Plant in MODE 3	Time 42 hours- 12 hour AOT Expires Plant in MODE 3

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.2 - JET PUMPS**

As depicted above, 42 hours is the maximum time that would be allowed if a required jet pump flow indicator is inoperable and jet pump flow can not be determined. Currently a maximum of 36 hours is allowed if more than one jet pump flow indicator is inoperable. Jet pump flow indication operability does not directly impact jet pump operability. Jet pump flow indication is only required to perform the jet pump Surveillance (SR 3.4.2.1). SR 3.4.2.1 verifies jet pump operability and has a frequency of every 24 hours. The 24 hours frequency plus the 25% extension has been shown by operating experience to be timely for detecting jet pump degradation and is consistent with the surveillance frequency for recirculation loop operability verification. The most common outcome of the performance of a surveillance is the successful demonstration that the acceptance criteria are satisfied. This change is consistent with NUREG-1433.

- L2 Note 1 allows this Surveillance not to be performed until 4 hours after the associated recirculation loop is in operation, since these checks can only be performed during jet pump operation. The 4 hours is an acceptable time to establish conditions appropriate for data collection and evaluation. Note 2 to proposed SR 3.4.2.1 provides time to perform the required Surveillance when the reactor exceeds 25% RTP. Below 25% RTP, low jet pump flow results in indication which precludes the collection of repeatable and meaningful data. The flexibility to proceed to $\geq 25\%$ RTP and then commence the SR every 24 hours is consistent with approved Technical Specifications for both Perry Nuclear Power Plant and River Bend Station.
- L3 The allowed difference between each jet pump diffuser-to-lower plenum differential pressure to the loop average has been increased to 20%. This change is consistent with the recommendations of SIL-330 and NUREG/CR-3052 (Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure). SIL-330 specifies a 10% criteria for individual jet pump flow distribution. When measured by jet pump diffuser-to-lower plenum differential pressure, the equivalent limit is 20% because of the relationship between flow and delta-P. Since BFN uses the diffuser-to-lower plenum differential pressure measurement, the variance allowed should be 20% as recommended by SIL-330 and NUREG/CR-3052. This is a relaxation from existing requirements, therefore, it constitutes a less restrictive change. This increase in allowed difference is considered an acceptable criterion for verifying jet pump operability and is consistent with the BWR Standard Technical Specifications, NUREG 1433.



UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



DEC 07 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

A1

~~SURVEILLANCE REQUIREMENTS~~

3.6.C Coolant Leakage

- Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

- If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

A1

~~3.6.D Relief Valves~~

Action A

- When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

L2

A1

Modes 1, 2 + 3
Applicability

Proposed Note to SR 3.4.3.2

A3

4.6.C Coolant Leakage

- With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

See Justification for Changes to BFN 1ST 3.4.4 + 3.4.5

A1

LAI

~~4.6.D Relief Valves~~
SR 3.4.3.1

- Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle.

A2

All 13 valves will have been checked or replaced upon the completion of every second cycle.

SR 3.4.3.2

- In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

A2

LAI



~~(A1) 3.6.4.6 PRIMARY SYSTEM BOUNDARY~~

NOV 18 1988

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.6.D. Relief Valves~~

3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.

A5

LA2

4. At least one relief valve shall be disassembled and inspected each operating cycle.

3.6.E. Jet Pumps

1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

- a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.
- b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.
- c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

See Justification for Changes to BFN ISTS 3.4.2, Jet Pumps



(A1)

~~1.2/2.2 REACTOR COOLANT SYSTEM INTEGRITY~~

SAPRTY LIMIT

1.2 Reactor Coolant System Integrity

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specifications

- A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

See Justification for Change to BFN ISTS 2.0

LIMITING SAFETY SYSTEM SETTING

2.2 Reactor Coolant System Integrity

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

Limiting Safety Protective Action System Setting
SR 3.4.3.1

- A. Nuclear system relief valves open--nuclear system pressure

1,105 psig ±

33.2 ± psi

(4 valves)

(L1)

1,115 psig ±

33.5 ± psi

(4 valves)

1,125 psig ±

33.8 ± psi

(5 valves)

- B. Scram--nuclear system high pressure

≤1,055 psig



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



DEC 07 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.6.C Coolant Leakage

- 2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

- 3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.C Coolant Leakage

- 2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

SEE JUSTIFICATION FOR CHANGES TO BFN ISTS 3.4.4 + 3.4.5

(A1) ~~3.6.D Relief Valves~~

- 1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

ACTION A

(L2)

(A1) MODES 1, 2 + 3
Applicability

(A3) Proposed Note to SR 3.4.3.2

(A1) ~~4.6.D Relief Valves~~

SR 3.4.3.1

- 1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

(A2)

- 2. In accordance with Specification 1.0.MI, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

SR 3.4.3.2

(LA1)



~~4.6.9. Relief Valves~~

A5

3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.

LA2

4. At least one relief valve shall be disassembled and inspected each operating cycle.

3.6.3. Jet Pumps

1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be shutdown in the COLD SHUTDOWN CONDITION within 24 hours.

5. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

- a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.
- b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.
- c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

See Justification for Changes to BFN ISTS 3.4.2, Jet Pumps

1.2/2.2 REACTOR COOLANT SYSTEM INTEGRITY

AI

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

1.2 Reactor Coolant System Integrity

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specifications

- A. The pressure at the lowest point of the reactor vessel shall not exceed 1.375 psig whenever irradiated fuel is in the reactor vessel.

SEE JUSTIFICATION FOR CHANGES TO BFN 1STS 2.0

2.2 Reactor Coolant System Integrity

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

Limiting Safety Protective Action System Setting

SR 3.4.3.1

- A. Nuclear system relief valves $1,105 \text{ psig} \pm 33.2 \text{ psi}$ open--nuclear system pressure (4 valves)

L1 $1,115 \text{ psig} \pm 33.5 \text{ psi}$ (4 valves)

$1,125 \text{ psig} \pm 33.8 \text{ psi}$ (5 valves)

- B. Scram- nuclear system high pressure $\leq 1,055 \text{ psig}$

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.C Coolant Leakage

- 2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

- 3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.C Coolant Leakage

- 2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

See Justification for changes to BFN 1STS 3.4.4 + 3.4.5

(A1)

~~3.6.D. Relief Valves~~

(A4)

be in mode 3 in 12hrs

Action A

- 1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

(L2)

(A1)

modes 1, 2 + 3

Applicability

Proposed Note to SR 3.4.3.2

(A3)

(M1)

A1

~~4.6.D. Relief Valves~~

SR 3.4.3.1

- 1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve, each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

(A2)

SR 3.4.3.2

- 2. In accordance with Specification 1.0.M1, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve

(LA2)

(LA1)



NOV 18 1988

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

A1

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~A.6.D. Relief Valves~~

A5 3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.

LA2 4. At least one relief valve shall be disassembled and inspected each operating cycle.

3.6.E. Jet Pumps

1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is INOPERABLE, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

See Justification for Changes to BFN ISTRS 3.4.2 Jetpumps

(A1)

Specification 3.4.3

~~1.2/2.2 REACTOR COOLANT SYSTEM INTEGRITY~~

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

1.2 Reactor Coolant System Integrity

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification

- A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

2.2 Reactor Coolant System Integrity

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

SR 3.4.3.1

- A. Nuclear system relief valves $1,105 \text{ psig} \pm 33.2$ psi open—nuclear (4 valves) system pressure

④ $1,115 \text{ psig} \pm 33.5$ psi (4 valves)

$1,125 \text{ psig} \pm 33.3$ psi (5 valves)

- B. Scram—nuclear $\leq 1,055 \text{ psig}$ system high pressure

See justification for charges to BFN ISTS 2.0

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.3 - SAFETY/RELIEF VALVES**

ADMINISTRATIVE

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The Frequency for proposed SR 3.4.3.1 (CTS 4.6.D.1) has been changed from "each operating cycle" to "18 months." Since an operating cycle is 18 months these are equivalent. The Frequency for proposed SR 3.4.3.2 (CTS 4.6.D.2) has been changed from "In accordance with Specification 1.0 MM" to "18 months." Since the Inservice Testing Program (1.0MM) frequency is 18 months these are equivalent. As such, these changes are considered administrative.
- A3 The proposed change adds a note that states that the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME code requirements, prior to valve installation. As such, the addition of the note is considered administrative.
- A4 CTS 3.6.D.1 requires an orderly shutdown when more than one relief valve is known to have failed. Therefore, the CTS allows unlimited operation with one S/RV inoperable. BFN has 13 S/RVs, therefore, 12 are required OPERABLE at all times. LCO 3.4.3 requires 12 to be OPERABLE and shutdown if one of the 12 required S/RVs is inoperable. As such, the two Specifications are equivalent and this change in presentation is considered administrative.

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.3 - SAFETY/RELIEF VALVES**

- A5 BFN CTS 4.6.D.3 is only applicable to three stage Target Rock S/RVs. Only the two stage Target Rock S/RVs are installed and authorized for use in BFN Unit 2. The three stage design is obsolete and is no longer supported at BFN. Since this Surveillance Requirement is no longer applicable to the BFN S/RV design, the deletion of this requirement is considered administrative.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 An additional requirement is being added that requires the plant to be in MODE 3 within 12 hours. This change is more restrictive because it stipulates that the reactor shutdown be completed much earlier than would be required by the existing specifications (CTS 3.6.D.1). CTS requires a shutdown to MODE 4 within 24 hours but does not stipulate how quickly MODE 3 must be reached. Reference Comment L2 which addresses the less restrictive change of be in MODE 4 in 36 hours rather than 24 hours.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

- LA1 The details relating to methods of performing Surveillances have been relocated to the Bases or procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the procedures will be controlled by the licensee controlled programs.
- LA2 This Surveillance Requirement has been relocated to plant procedures since the requirement does not directly relate to S/RV operability. This is strictly a preventive maintenance requirement.

PAGE 2 OF 3

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.3 - SAFETY/RELIEF VALVES

"Specific"

- L1 The allowed lift setpoint tolerance has been increased from 1% to 3% based on incorporation of this larger setpoint tolerance in the BFN reload licensing analysis for each Unit prior to ISTS implementation. The larger setpoint tolerance has already been incorporated into the Unit 2 reload analysis and will be incorporated into the Unit 3 reload analysis for the next cycle (Spring 1997). In addition, when the setpoints are verified, they are still required to be reset to 1% (proposed SR 3.4.3.1). Thus, since the analysis still ensure that all limits are maintained even with the expanded tolerance, this change is considered acceptable. This change is also consistent with the BWR Standard Technical Specifications, NUREG 1433.
- L2 The time to reach MODE 4 (reactor depressurized to < 105 psig, Cold Shutdown) has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added (Reference Comment M4 above). These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.

UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



AUG 26 1987

~~3.0/4.0 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

~~3.6.C. Coolant Leakage~~

Modes 1, 2 + 3

(A1)

~~4.6.G. Coolant Leakage~~

SR 3.4.4.1

1. a. **Applicability** Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

LCO 3.4.4.b

LCO 3.4.4.c

(M3)

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

(LA1)

(L2)

(12)

b. Anytime the reactor is in RUN MODE, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN MODE except as defined in 3.6.C.1.c below.

LCO 3.4.4.d

(A2)

Within the previous

(A2)

LCO 3.4.4.d

c. During the first 24 hours in the RUN MODE following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

(M1)

Add LCO 3.4.4.a

DEC 07 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1) ~~3.6.C Coolant Leakage~~

(A1) ~~4.6.C Coolant Leakage~~

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

See Justification for changes to BFN ISTS 3.4.5

(L3)

Add Action-A + Required Action B.1

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

ACTION C
(1st Condition)

Add Required Action B.2

Add 2nd Condition of Action C

(L5)

(M1)

3.6.D Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

Hot Shutdown Condition in 12 hours and

(L4)

(M2)

4.6.D Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

See Justification for changes to BFN ISTS 3.4.3 FN This Section

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

AUG 26 1987

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1) ~~3.6.6. Coolant Leakage~~

Modes 1, 2+3

Applicability

1. a.

Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

LCO 3.4.4.b

LCO 3.4.4.c

b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below.

LCO 3.4.4.d

(A2) within the previous

c. During the first 24 hours in the RUN mode following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

LCO 3.4.4.d

(M1) Add LCO 3.4.4.e

(A1) ~~3.6.6. Coolant Leakage~~

SR 3.4.4.1

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

(LAI)

(L2) 12

(L1)

DEC 07 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

(A1) ~~3.6.C Coolant Leakage~~

(A1) ~~4.6.C Coolant Leakage~~

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

SEE JUSTIFICATION FOR CHANGES TO BFN ISTS 3.4.5 IN THIS SECTION

(L3) Add ACTION A + Required Action B.1

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

ACTION C (1st Condition)

Add Required Action B.2

Add 2nd Condition of ACTION C (M1)

(L4) 3.6.D Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

HOT SHUTDOWN CONDITION in 12 hours and

4.6.D Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.FM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

SEE JUSTIFICATION FOR CHANGES TO BFN ISTS 3.4.3 IN THIS SECTION



UNIT 3
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

AUG 26 1987

~~3.6/A.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

~~3.6.C Coolant Leakage~~

Modes 1, 2+3

Applicability

1. a.

Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

LCO 3.4.4.b

LCO 3.4.4.c

b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below.

LCO 3.4.4.d

(A2)

within the previous

c. During the first 24 hours in the RUN mode following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

(A2)

LCO 3.4.4.d

(M1) Add LCO 3.4.4.a →

(A1)

~~3.6.C Coolant Leakage~~

SR 3.4.4.1

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

(L1)

(L2)

12

(L1)

(30)

~~2.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

~~3.6.C Coolant Leakage~~

(A1)

~~4.6.C Coolant Leakage~~

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

See Justification for Changes to BFN ISTS 3.4.5 In this Section

(L3)

ADD ACTION A + Required Action B.1

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

ACTION C (1st Condition)

(L4)

3.6.D. Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

HOT SHUTDOWN Condition in 12 hours and

(M2)

See Justification for Changes to BFN ISTS 3.4.3 In this Section

(L5)

ADD Required Action B.2

ADD 2nd Condition of Action C (M1)

4.6.D. Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.4 - RCS OPERATIONAL LEAKAGE**

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 The total LEAKAGE limit applies at any moment, to the previous 24 hours (not any future or past 24 hour period). This results in a "rolling average" covering "any 24-hour period." Therefore, changing "any" to "the previous" does not change any intent. In addition, the current provision (CTS 3.6.C.1.c), which allows an increase in reactor coolant leakage into the primary containment of >2 gpm during the first 24 hours in the RUN mode following STARTUP as long as unidentified leakage and total leakage limits are not exceeded, is encompassed by proposed LCO 3.4.4.d which allows the same. LCO 3.4.4.d is worded differently (i.e., ≤ 2 gpm increase in unidentified leakage within the previous 24 hour period in MODE 1) but means the same. Since there is no "previous" 24 hour period until being in MODE 1 for 24 hours, this limit does not apply for the first 24 hours. These are editorial changes only and as such are considered administrative.

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A new requirement has been added to preclude pressure boundary LEAKAGE. An applicable ACTION has also been added. This is an additional restriction on plant operation.

PAGE 1 OF 4



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.4 - RCS OPERATIONAL LEAKAGE

M2 CTS 3.6.C.3 requires an orderly shutdown be initiated and the reactor to be in the COLD SHUTDOWN CONDITION within 24 hours when certain conditions can not be met. Proposed Action C will require the plant be in MODE 3 in 12 hours and MODE 4 in 36 hours. The addition of this intermediate step to the COLD SHUTDOWN CONDITION is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.

M3 The proposed applicability of MODES 1, 2 and 3 is more restrictive than CTS 3.6.C.1.a applicability of "Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F." The Startup Mode will now include the mode switch position of "Refuel" when the head bolts are fully tensioned. The change eliminates the potential to interpret certain plant conditions such that no MODE, or a less restrictive MODE, would exist. Currently, CTS 1.0.M allows the plant to be considered in the SHUTDOWN CONDITION and in the Shutdown Mode with the mode switch in the Refuel position (and other positions are allowed while in the Shutdown Mode) as permitted by notes to that definition.

The allowance to place the Mode Switch in other positions has been moved to Section 3.10, Special Operations and Section 3.3.2.1, Control Rod Block Instrumentation. Any technical changes to these allowances will be discussed in the Justification for Changes to these Sections.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LAI Details of the methods for performing this Surveillance are relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the procedures will be controlled by the licensee controlled programs.

PAGE 2 OF 4



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.4 - RCS OPERATIONAL LEAKAGE

"Specific"

- L1 The total LEAKAGE allowed has been increased to 30 gpm. No applicable safety analysis assumes the total LEAKAGE limit. The limit considers RCS inventory makeup and drywell floor drain capacity. The new limit of 30 gpm is well within the capacity of the Control Rod Drive System pump and the RCIC System, and is well below the capacity of one drywell equipment drain or floor drain pump, which is used to pump the water out of the collecting sump. The collecting sumps can also accommodate this small additional leakage rate.
- L2 The Frequency has been changed from 4 hours to 12 hours, consistent with the allowance in Generic Letter 88-01, Supplement 1. The supplement allows the Frequency to be extended to shiftly, not to exceed 12 hours. Browns Ferry Technical Specifications currently define the frequency of shiftly as 12 hours, thus, this Frequency is adjusted to coincide with this.
- L3 CTS do not provide a period of time to reduce leakage prior to initiating an orderly shutdown. Proposed ACTIONS A and B allow 4 hours to reduce LEAKAGE within limits prior to initiating a shutdown. This is reasonable since the total leakage limits are conservatively below the LEAKAGE that would constitute a critical crack size. The 4 hour completion time for ACTION B is reasonable to properly verify the source of unidentified leakage before the reactor must be shutdown without unduly jeopardizing plant safety. The proposed changes are consistent with the BWR/4 Standard Technical Specifications, NUREG 1433.
- L4 The time allowed to shutdown the plant when the required actions are not met has been changed from "in the COLD SHUTDOWN CONDITION within 24 hours" to in MODE 3 (Hot Shutdown) in 12 hours and MODE 4 (Cold Shutdown) within 36 hours. The proposed allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The additional 12 hours allowed to reach Mode 4 is offset by the safety benefit of being subcritical (MODE 3) in a shorter required time.

PAGE 3 OF 4



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.4 - RCS OPERATIONAL LEAKAGE**

- L5 Proposed LCO 3.4.4, RCS Operational Leakage, will add an alternative to existing requirement in Specifications 3.6.C.1 and 3.6.C.3 that a reactor shutdown be initiated if unidentified leakage increases at a rate of more than 2 gpm within a 24 hour period. Under proposed Required Action B.2, unidentified leakage that increases at a rate of more than 2 gpm within a 24 hour period will not require initiation of a reactor shutdown if it can be determined within 4 hours that the source of the unidentified leakage is not service sensitive type 304 and type 316 austenitic stainless steel piping that is subject to high stress or that contains relatively stagnant or intermittent flow fluids. This alternative Required Action is acceptable because the low limit on the rate of increase of unidentified leakage was established as a method for early identification of Intergranular Stress Corrosion Cracking (IGSCC) in Type 304 and Type 316 austenitic stainless steel piping. IGSCC produces tight cracks and the small flow increase limit is capable of providing an early warning of such deterioration. Verification that the source of leakage is not Type 304 and Type 316 austenitic stainless steel eliminates IGSCC as a cause of leak. This significantly reduces concerns about crack instability and the rapid failure in the RCS boundary. Also, the unidentified LEAKAGE limit is still being maintained and will continue to limit the maximum unidentified LEAKAGE allowed. This change is consistent with NUREG-1433.

PAGE 4 OF 4



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



DEC 07 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1) ~~3.6.C Coolant leakage~~

(Modes 1, 2 + 3)
(M1)

(A1) ~~4.6.C Coolant Leakage~~

Applicability

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

LCD 3.4.6

ACTIONS A+B

Proposed Note to Actions A+B

Required Action B.1

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

(24)
(12)

(A4)

(A6)

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

(L4)
30 days

(M2)

Actions C+D

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(36)
(L1)

(A2)

(M6)

the HOT SHUTDOWN CONDITION in 12 hours and

3.6.D Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

4.6.D Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

See Justification for Changes to BFN ISTS 3.4.3 in this section



TABLE 3.2.E
INSTRUMENTATION THAT MONITORS LEAKAGE INTO DRYWELL

System (2)	Setpoints	Action	Remarks
Equipment Drain Flow Integrator Sump Fill Rate Timer Sump Pump Out Rate Timer	N/A >20.1 min. <13.4 min.	(1)	1. Used to determine identifiable reactor coolant leakage. 2. Considered part of sump system.
Floor Drain Flow Integrator Sump Fill Rate Timer Sump Pump Out Rate Timer	N/A >80.4 min. <8.9 min.	(1)	1. Used to determine unidentifiable reactor coolant leakage. 2. Considered part of sump system.
LCD 3.4.5.a LCD 3.4.5.b Drywell Air Sampling	L3 Or Gas and Particulate 3 x Average Background	(3)	

NOTES:

- (1) Whenever a system is required to be operable, there shall be one operable system either automatic or manual, or ³ ACTION A
- (2) ~~An alternate system to determine the leakage flow is a manual system whereby the time between sump pump starts is monitored. The time interval will determine the leakage flow because the volume of the sump will be known.~~ LA1
- (3) Upon receipt of alarm, immediate action will be taken to confirm the alarm and assess the possibility of increased leakage. LA3

BFW Unit 1



TABLE 4.2.E
 MINIMUM TEST AND CALIBRATION FREQUENCY FOR DRYWELL LEAK DETECTION INSTRUMENTATION

A3

SR 3.4.5.1

Function	Functional test SR 3.4.5.3	Calibration SR 3.4.5.4	Instrument Check SR 3.4.5.1
LAS → Equipment Drain Sump Flow Integrator	(1)	once/6 months	once/day
Floor Drain Sump Flow Integrator	(4) LA4	once/6 months (184 days) AS	once/day L2
Air Sampling System SR 3.4.5.2 →	(1) 31 days AS	once/3 months (18 months) LS	once/day
LAS → Equipment Drain Sump Fill Rate and Pumpout Rate Timers	(4)	once/operating cycle	N/A (12 hrs) M5
LA4 → Floor Drain Sump Fill Rate and Pumpout Rate Timers	(4)	once/operating cycle	N/A
LAS → Equipment Drain Logic	once/operating cycle	(6)	N/A
LA4 → Floor Drain Logic	once/operating cycle	(6)	N/A

3.2/4.2-53

BFN Unit 1

FEB 05 1987

Specification 3.4.5

only Notes 1, 4 and 6 apply to Table 4.2.E. The remaining Notes are addressed in the markups to Section 3.3 Instrumentation.

Justification 3.7.3

JAN 26 1989

(A1)

~~NOTES FOR TABLES 4.2.A THROUGH 4.2.L except 4.2.D AND 4.2.E~~
SR 3.4.5.2 Free

31 days

(A5)

1. Functional tests shall be performed once per month.
2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.
3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

~~4. Tested during logic system functional tests.~~

(LA4)

5. Refer to Table 4.1.B.

~~6. The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.~~

7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.
8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be OPERABLE or are tripped.
9. Calibration frequency shall be once/year.
10. Deleted
11. Portion of the logic is functionally tested during outage only.
12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.
13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

See Justification for Changes for BFN ISTS 3.3

BFN
Unit 1

3.2/4.2-59

AMENDMENT NO. 164



UNIT 2
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

(A1)

~~3.6.6 Coolant Leakage~~

Modes 1, 2+3

(M1)

Applicability

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

LCO 3.4.6

ACTIONS A+B

Proposed Note to ACTION B

(A1)

(A6)

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

30 days (L4)

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

ACTIONS C+D

(A2)

(M6)

the HOT SHUTDOWN CONDITION in 12 hours and

(L1)

~~3.6.D Relief Valves~~

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

SEE JUSTIFICATION FOR CHANGES TO BFN ISTS 2.4.3 IN THIS SECTION

(A1)

~~4.6.6 Coolant Leakage~~

Required Action B.1

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

12

(M2)

~~4.6.D Relief Valves~~

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.



TABLE 3.2.E
INSTRUMENTATION THAT MONITORS LEAKAGE INTO DRYWELL

System (2)	Setpoints	Action	Remarks
Equipment Drain Flow Integrator Sump Fill Rate Timer Sump Pump Out Rate Timer	N/A ≥20.1 min. ≤13.4 min.	(1)	1. Used to determine identifiable reactor coolant leakage. 2. Considered part of sump system.
Floor Drain Flow Integrator Sump Fill Rate Timer Sump Pump Out Rate Timer	N/A ≥80.4 min. ≤8.9 min.	(1)	1. Used to determine unidentifiable reactor coolant leakage. 2. Considered part of sump system.
Drywell Air Sampling	Gas and Particulate 3 X Average Background	(3)	

LCO 3.4.5.a

LCO 3.4.5.b
3.2/4.2-30

NOTES:

- (1) Whenever a system is required to be operable, there shall be one operable system either automatic or manual, or the action required in Section 3.6.C.2 shall be taken. } ACTION A
- (2) An alternate system to determine the leakage flow is a manual system whereby the time between sump pump starts is monitored. The time interval will determine the leakage flow because the volume of the sump will be known. LA1
- (3) Upon receipt of alarm, immediate action will be taken to confirm the alarm and assess the possibility of increased leakage. LA3

BFN-Unit 2

TABLE 4.2.E
 MINIMUM TEST AND CALIBRATION FREQUENCY FOR DRYWELL LEAK DETECTION INSTRUMENTATION

A3

SR 3.4.5.1

Function	Functional Test	SR 3.4.5.3 Calibration	SR 3.4.5.4 Instrument Check
LA5 Equipment Drain Sump Flow Integrator	(4)	once/6 months	once/day
Floor Drain Sump Flow Integrator	(4) LA4	once/6 months AS	once/day L2
Air Sampling System SR 3.4.5.2	(4) 31 days AS	once/3 months AS	once/day 12 hrs L5
LA5 Equipment Drain Sump Fill Rate and Pumpout Rate Timers	(4)	once/operating cycle	N/A MS
LA4 Floor Drain Sump Fill Rate and Pumpout Rate Timers	(4)	once/operating cycle	N/A
LA5 Equipment Drain Logic	once/operating cycle	(6)	N/A
LA4 Floor Drain Logic	once/operating cycle	(6)	N/A

3.2/4.2-53

BFN-Unit 2



Only Notes 1, 4 and 6 apply to Table 4.2.E. The remaining Notes are addressed in the manuscripts to Section 3.3, Instrumentation.

Specification 34.5

(A1)

~~NOTES FOR TABLES 4.2.A THROUGH 4.2.I except 4.2.D AND 4.2.K~~

JAN 26 1989

SR 34.5.2 Freq.

1. Functional tests shall be performed once per ~~month~~ ^{31 days} (AS).
2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.
3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.
4. Tested during logic system functional tests. (LAY)
5. Refer to Table 4.1.B.
6. The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.
7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.
8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be OPERABLE or are tripped.
9. Calibration frequency shall be once/year.
10. Deleted
11. Portion of the logic is functionally tested during outage only.
12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.
13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

See Justification for Changes
for BFN 1STS 3.3

UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

~~2.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.C Coolant Leakage~~ (Modes 1, 2, & 3) (m)

~~4.6.C Coolant Leakage~~ (A1)

Applicability 2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

Required Action B.1

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

LCD 3.4.6

ACTIONS A+B

Proposed Note to Actions A+B

(A4)

(A6)

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

30 days

(L4)

ACTIONS C+D

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

the HBT SHUTDOWN CONDITION in 12 hours and

(A2)

(M6)

3.6.D. Relief Valves (L1)

4.6.D. Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve

See Justification for CHANGES TO BEN 1STS 3.4.3 in this Section



TABLE 3.2.E
INSTRUMENTATION THAT MONITORS LEAKAGE INTO DRYWELL

(A3)

System (2)	Setpoints	Action	Remarks
Equipment Drain Flow Integrator Sump Fill Rate Timer Sump Pump Out Rate Timer	N/A ≥20.1 min. ≤13.4 min.	(1)	1. Used to determine identifiable reactor coolant leakage. 2. Considered part of sump system.
Floor Drain Flow Integrator Sump Fill Rate Timer Sump Pump Out Rate Timer	N/A ≥60.4 min. ≤8.9 min.	(1)	1. Used to determine unidentifiable reactor coolant leakage. 2. Considered part of sump system.
LCO 3.4.5.a LCO 3.4.5.b Drywell Air Sampling	2 of Gas and Particulate 3 x Average Background	(3)	

(LA5)

(A3)

LCO
3.4.5.a

LCO
3.4.5.b Drywell Air Sampling

3.2/4.2-29

(LA2)
N/A
≥60.4 min.
≤8.9 min.
2 of Gas and Particulate
3 x Average Background

(3)

NOTES:

(1) Whenever a system is required to be operable, there shall be one operable system either automatic or manual, or the action required in Section 3.6.C.2 shall be taken.

Action A

(2) An alternate system to determine the leakage flow is a manual system whereby the time between sump pump starts is monitored. The time interval will determine the leakage flow because the volume of the sump will be known.

(LA1)

(3) Upon receipt of alarm, immediate action will be taken to confirm the alarm and assess the possibility of increased leakage.

(LA3)

BFN-Unit 3



TABLE 4.2.E
MINIMUM TEST AND CALIBRATION FREQUENCY FOR DRYWELL LEAK DETECTION INSTRUMENTATION

A3

SR 3.4.5.1

Function	Functional Test SR 3.4.5.3	Calibration SR 3.4.5.4	Instrument Check SR 3.4.5.1
LA5 Equipment Drain Sump Flow Integrator	(6)	once/6 months	once/day
Floor Drain Sump Flow Integrator	(4) LA4	once/6 months AS 187 days	once/day L2
Air Sampling System SR 3.4.5.2	(1) 31 days AS	18 once/3 months LS	once/day 12hrs MS
LA5 Equipment Drain Sump Fill Rate and Pumpout Rate Timers	(4)	once/operating cycle	N/A
LA4 Floor Drain Sump Fill Rate and Pumpout Rate Timers	(4)	once/operating cycle	N/A
LA5 Equipment Drain Logic	once/operating cycle	(6)	N/A
LA4 Floor Drain Logic	once/operating cycle	(6)	N/A

3.2/4.2-52

BFM-Unit 3



Only Notes 1, 4 and 6 apply to Table 4.2.E. The remaining notes are addressed in the markups to Section 3.3, Instrumentation.

Specification 3.4.5

(A1)

~~NOTES FOR TABLES 4.2.A THROUGH 4.2.I except 4.2.D AND 4.2.K~~

JAN 26 1989

SR 3.4.5.2 Free

1. Functional tests shall be performed once per ~~month~~ ^{31 days} (45)

2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. ~~Tested during logic system functional tests.~~

LA4

5. Refer to Table 4.1.B.

6. ~~The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.~~

7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.

8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be operable or are tripped.

9. Calibration frequency shall be once/year.

10. (DELETED)

11. Portion of the logic is functionally tested during outage only.

12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.

13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

see Justification for Changes
for BFN ISTS 3.3



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 The revised presentation of actions is proposed to explicitly identify that LCO 3:0.3 is required to be entered if all required RCS leakage monitoring systems are inoperable. This action is consistent with the current requirements and is considered a presentation preference. Therefore, this change is considered administrative.

A3 The Table format is being deleted. This change is considered a presentation preference. Therefore, this change is considered administrative.

A4 Proposed ACTION B is modified by a note that explicitly states that the provisions of 3.0.4 are not applicable. This explicitly allows a mode change when both the particulate and gaseous primary containment monitoring channels are inoperable. This allowance is provided because, in this Condition, the drywell sump monitoring system will be available to monitor RCS leakage and the compensatory actions for the inoperable system will provide additional indication of RCS leakage. This is an administrative change since existing Technical Specifications do not have an explicit requirement that prohibits entry into a Mode or condition when an LCO required by that Mode or condition is not satisfied. Therefore, CTS allows the actions being permitted by the note being added. This is consistent with NUREG-1433.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

- A5 Frequency has been editorially changed from monthly to every 31 days and from every six months to every 184 days. This is an administrative change since these are equivalent time periods.
- A6 The current provision (CTS 3.6.C.2, 2nd paragraph) that allows the air sampling system to be removed from service for a period of 4 hours for calibration, functional testing, and maintenance without providing a temporary monitor has been eliminated. There is currently no requirement for a monitor for at least 24 hours (CTS 4.6.C.2). Therefore, the current provision serves no purpose.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 The proposed applicability of MODES 1, 2 and 3 is more restrictive than CTS 3.6.C.1.a applicability of "Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F." The Startup Mode will now include the mode switch position of "Refuel" when the head bolts are fully tensioned. The change eliminates the potential to interpret certain plant conditions such that no MODE, or a less restrictive MODE, would exist. Currently, CTS 1.0.M allows the plant to be considered in the SHUTDOWN CONDITION and in the Shutdown Mode with the mode switch in the Refuel position (and other positions are allowed while in the Shutdown Mode) as permitted by notes to that definition.

The allowance to place the Mode Switch in other positions has been moved to Section 3.10, Special Operations and Section 3.3.2.1, Control Rod Block Instrumentation. Any technical changes to these allowances will be discussed in the Justification for Changes to these Sections.

- M2 The frequency of grab sampling with the air sampling system inoperable has been increased from 24 hours to 12 hours. A grab sample once/12 hours provides adequate information to detect leakage during the extended (See Justification for Change L4) period of time that the air sampling system is allowed to be inoperable.
- M3 Not used.
- M4 Not used.
- M5 The Frequency of the channel check requirement has been changed from every 24 hours to every 12 hours, consistent with Generic Letter 88-01, Supplement 1 and NUREG-1433. This is an additional restriction on plant operation.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

M6 CTS 3.6.C.3 requires an orderly shutdown be initiated and the reactor to be in the COLD SHUTDOWN CONDITION within 24 hours when certain conditions can not be met. Proposed Action C will require the plant be in MODE 3 in 12 hours and MODE 4 in 36 hours. The addition of this intermediate step to the COLD SHUTDOWN CONDITION is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

- LA1 The description of an acceptable alternate system to measure leakage has been relocated to the Bases or procedures that support compliance with the limits for RCS Operational Leakage in proposed Specification 3.4.4. The design features and system operation are also described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the procedures and FSAR will be controlled by the provisions of 10 CFR 50.59.
- LA2 The details relating to the setpoints have been relocated to the procedures. Changes to the procedures will be controlled by the licensee controlled programs.
- LA3 The details relating to actions required upon receipt of an alarm have been relocated to procedures. Changes to the procedures will be controlled by the licensee controlled programs.
- LA4 Details of the specifics of the functional, calibration, and logic system functional test related to the floor drain sump fill rate and pump out timers has been relocated to procedures since the operability of the system is not dependent upon these timers. Changes to the procedures will be controlled by the licensee controlled programs.



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION**

- LA5 The drywell equipment drain sump monitoring system functions to quantify identified leakage. Since the purpose of this specification is to provide early indication of unidentified RCS leakage, the drywell equipment drain sump monitoring system has been relocated to the Bases or procedures that support compliance with the limits for RCS Operational Leakage in proposed Specification 3.4.4. The design features and system operation are also described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the procedures and FSAR will be controlled by the provisions of 10CFR50.59.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Specific"

- L1 The time allowed to shutdown the plant when the required actions are not met has been changed from "in the COLD SHUTDOWN CONDITION within 24 hours" to in MODE 3 (Hot Shutdown) in 12 hours and MODE 4 (Cold Shutdown) within 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added. These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.
- L2 This requirement has been deleted. An instrument check would not consistently demonstrate operability since normally the instruments could not be compared to any other instruments, and their reading could be anywhere on scale; thus, observing the meter would provide no valid information as to whether the instrument is OPERABLE. The CHANNEL FUNCTIONAL TEST requirement is the best indicator of OPERABILITY while operating, and this requirement is being maintained. This is also consistent with the BWR Standard Technical Specification, NUREG 1433.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

- L3 CTS Table 3.2.E defines the air sampling system as consisting of gas and particulate monitoring channels (i.e., both channels are required OPERABLE for the air sampling system to be considered OPERABLE). Proposed LCO 3.4.5.b requires either one channel of the gas or one channel of the particulate monitoring system to be OPERABLE. This is less restrictive than CTS requirements but is acceptable since either channel is capable of indicating increased LEAKAGE rates that correlate to radioactivity levels of 3 times average background.
- L4 The allowed outage time for the air sampling system has been changed from 72 hours to 30 days. The 30 day allowed outage time recognizes that at least one other form of leak detection is available (sump monitoring) and takes credit for the increased sampling frequency of 12 hours (versus CTS of 24 hrs). This change is consistent with NUREG-1433.
- L5 The calibration frequency has been changed once per 3 months to once per 18 months. This new Frequency is consistent with BFN setpoint methodology, which considers the magnitude of the equipment drift in the setpoint analysis over an 18 month calibration interval. The primary containment leak detection noble gas and particulate monitor is a digital Eberline continuous air monitor (CAM) which is identical to the building effluent monitors whose calibration frequency is 18 months in accordance with the Offsite Dose Calculation Manual (ODCM) and previously required by Technical Specification Table 4.2.K until these instruments were removed by Amendment No. 216 dated September 22, 1993 (reference TS 301). Excessive calibration can cause damage to the equipment. In addition, plant operations could be impacted while the equipment is removed from service for calibration since it would not be available for leak detection.



UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.B. Coolant Chemistry

4. When the reactor is not pressurized with fuel in the reactor vessel, except during the STARTUP CONDITION, the reactor water shall be maintained within the following limits.

- a. Conductivity - 10 μ mho/cm at 25°C
- b. Chloride - 0.5 ppm
- c. pH shall be between 5.3 and 8.6.

5. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated immediately. The reactor shall be brought to the COLD SHUTDOWN CONDITION as rapidly as cooldown rate permits.

LCO
3.4.6
+
Applic

6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 μ Ci/gm of dose equivalent I-131.

(M1)

4.6.B. Coolant Chemistry

4. Whenever the reactor is not pressurized with fuel in the reactor vessel, a sample of the reactor coolant shall be analyzed at least every 96 hours for conductivity, chloride ion content and pH.

See Justification for Changes for CTS 3.6.8/4.6.8 in this Section

Note for SR 3.4.9.1

(M2)

SR 3.4.6.1

5. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least I-131, I-132, I-133, and I-134 shall be performed monthly on a coolant liquid sample.

(LAI)

(M3)

7 days

Required Action A.1+B.1

6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:

(LAI)



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.B. Coolant Chemistry~~ (A2)

~~4.6.B. Coolant Chemistry~~

3.6.B.6 (Cont'd) Proposed Note to Required Actions for Condition A

~~4.6.B.6 (Cont'd)~~

ACTION A

(L1)

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 $\mu\text{Ci/gm}$ whenever the reactor is critical. The reactor shall not be operated more than 5% of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 $\mu\text{Ci/gm}$, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

ACTION B

(L2) within 12 hours

(A3) or be in Mode 4 within 36 hours

(LAI)

(MI)

Rea Act A.1

(LAI)

- a. During the STARTUP CONDITION
- b. Following a significant power change**
- c. Following an increase in the equilibrium off-gas level exceeding 10,000 $\mu\text{Ci/sec}$ (at the steam jet air ejector) within a 48-hour period

d. Whenever the equilibrium iodine limit specified in 3.6.B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 $\mu\text{Ci/gm}$) is established. However, at least 3 consecutive samples shall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration.

7. When there is no fuel in the reactor vessel, technical specification reactor coolant chemistry limits do not apply.

7. When there is no fuel in the reactor vessel, sampling of reactor coolant chemistry at technical specification frequency is not required.

See Justification for Changes for CTS 3.6.B/4.6.B in this Section

** For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.

(LAI)



(M1)

INSERT PROPOSED NEW SPECIFICATION 3.4.7

Insert new Specification 3.4.7, Residual Heat Removal System - Hot Shutdown, as shown in the BFN Unit 2 Improved Technical Specifications.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.7
RHR SHUTDOWN COOLING SYSTEM - HOT SHUTDOWN

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 A new Specification is being added requiring two RHR Shutdown Cooling subsystems to be OPERABLE in MODE 3 with reactor steam dome pressure less than the RHR low pressure permissive pressure. Appropriate ACTIONS and a Surveillance Requirement are also added. This is consistent with the BWR Standard Technical Specification, NUREG 1433 and is an additional restriction on plant operation.

(MII)

INSERT PROPOSED NEW SPECIFICATION 3.4.8

Insert new Specification 3.4.8, Residual Heat Removal System - Cold Shutdown, as shown in the BFN Unit 2 Improved Technical Specifications.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.8
RHR SHUTDOWN COOLING SYSTEM - COLD SHUTDOWN

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A new Specification is being added requiring two RHR Shutdown Cooling subsystems to be OPERABLE in MODE 4. Appropriate ACTIONS and a Surveillance Requirement are also added. This is consistent with the BWR Standard Technical Specification, NUREG 1433 and is an additional restriction on plant operation.



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

JUN 28 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.6.B. Coolant Chemistry

4. When the reactor is not pressurized with fuel in the reactor vessel, except during the STARTUP CONDITION, the reactor water shall be maintained within the following limits.

- a. Conductivity - 10 μ mho/cm at 25°C
- b. Chloride - 0.5 ppm
- c. pH shall be between 5.3 and 8.6.

5. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated immediately. The reactor shall be brought to the COLD SHUTDOWN CONDITION as rapidly as cooldown rate permits.

6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 μ Ci/gm of dose equivalent I-131.

LCO
3.4.6
+
Applic.

(M1)

4.6.B. Coolant Chemistry

4. Whenever the reactor is not pressurized with fuel in the reactor vessel, a sample of the reactor coolant shall be analyzed at least every 96 hours for conductivity, chloride ion content and pH.

See Justification for Changes for CTS 3.6.B/4.6.B in this section

(M2)

Note for SR 3.4.9.1

SR 3.4.6.1

5. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least I-131, I-132, I-133, and I-134 shall be performed monthly on a coolant liquid sample.

(LAI)

(M3)

every 7 days

Required Action A.1 + B.1

6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:

(LAI)



JUN 28 1994

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.B. Coolant Chemistry

4.6.B. Coolant Chemistry

3.6.B.6 (Cont'd)

4.6.B.6 (Cont'd)

Proposed Note to Required Action for Condition A

ACTION A

L1

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 $\mu\text{Ci/gm}$ whenever the reactor is critical. The reactor shall not be operated more than 5% of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 $\mu\text{Ci/gm}$, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

ACTION B

L2 within 12 hours

A3 or be in MUDS 4 with 36 hours

A1

LAI

MI

Req. Act. A.1

LAI

- a. During the STARTUP CONDITION
- b. Following a significant power change**
- c. Following an increase in the equilibrium off-gas level exceeding 10,000 $\mu\text{Ci/sec}$ (at the steam jet air ejector) within a 48-hour period.
- d. Whenever the equilibrium iodine limit specified in 3.6.B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 $\mu\text{Ci/gm}$) is established. However, at least 3 consecutive samples shall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration.

7. When there is no fuel in the reactor vessel, technical specification reactor coolant chemistry limits do not apply.

7. When there is no fuel in the reactor vessel, sampling of reactor coolant chemistry at technical specification frequency is not required.

See Justification for Changes for CTS 3.6B/4.6.B in this Section

LAI

** For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.



(M1)

INSERT PROPOSED NEW SPECIFICATION 3.4.7

Insert new Specification 3.4.7, Residual Heat Removal System - Hot Shutdown, as shown in the BFN Unit 2 Improved Technical Specifications.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.7
RHR SHUTDOWN COOLING SYSTEM - HOT SHUTDOWN

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A new Specification is being added requiring two RHR Shutdown Cooling subsystems to be OPERABLE in MODE 3 with reactor steam dome pressure less than the RHR low pressure permissive pressure. Appropriate ACTIONS and a Surveillance Requirement are also added. This is consistent with the BWR Standard Technical Specification, NUREG 1433 and is an additional restriction on plant operation.



(M1)

INSERT PROPOSED NEW SPECIFICATION 3.4.8

Insert new Specification 3.4.8, Residual Heat Removal System - Cold Shutdown, as shown in the BFN Unit 2 Improved Technical Specifications.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.8
RHR SHUTDOWN COOLING SYSTEM - COLD SHUTDOWN

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A new Specification is being added requiring two RHR Shutdown Cooling subsystems to be OPERABLE in MODE 4. Appropriate ACTIONS and a Surveillance Requirement are also added. This is consistent with the BWR Standard Technical Specification, NUREG 1433 and is an additional restriction on plant operation.

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



(A1)

3.6.B. Coolant Chemistry

4. When the reactor is not pressurized with fuel in the reactor vessel, except during the STARTUP CONDITION, the reactor water shall be maintained within the following limits.

- a. Conductivity - 10 μ mho/cm at 25°C
- b. Chloride - 0.5 ppm
- c. pH shall be between 5.3 and 8.6.

5. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated immediately. The reactor shall be brought to the COLD SHUTDOWN CONDITION as rapidly as cooldown rate permits.

6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 μ Ci/gm of dose equivalent I-131.

LCO
3.4.6
+
Applicability

(M1)

4.6.B. Coolant Chemistry

4. Whenever the reactor is not pressurized with fuel in the reactor vessel, a sample of the reactor coolant shall be analyzed at least every 96 hours for conductivity, chloride ion content and pH.

See Justification for Changes for CTS 3.6.B/4.6.B in this section

(M2)

Note for SR 3.4.9.1

SR 3.4.6.1

5. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least I-131, I-132, I-133, and I-134 shall be performed monthly on a coolant liquid sample.

(LAI)

(M3) every 7 days

Required Action A.1 & B.1

6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:

(LAI)



3.6.B. Coolant Chemistry

4.6.B. Coolant Chemistry

3.6.B.6 (Cont'd)

4.6.B.6 (Cont'd)

Proposed Note to Required Actions for Condition A

Action A

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 µCi/gm whenever the reactor is critical. The reactor shall not be operated more than 5% of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 µCi/gm, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

L1

LAI

M1

Action B

If the iodine concentration in the coolant exceeds 26 µCi/gm, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

L2 - Within 12 hours

A3 - or be in Mode 4 within 36 hours

Req Action A.1

LAI

- a. During the STARTUP CONDITION
- b. Following a significant power change**
- c. Following an increase in the equilibrium off-gas level exceeding 10,000 µCi/sec (at the steam jet air ejector) within a 48-hour period.
- d. Whenever the equilibrium iodine limit specified in 3.6:B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 µCi/gm) is established. However, at least 3 consecutive samples shall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration.

7. When there is no fuel in the reactor vessel, technical specification reactor coolant chemistry limits do not apply.

7. When there is no fuel in the reactor vessel, sampling of reactor coolant chemistry at technical specification frequency is not required.

See Justification for Changes for CTS 3.6.B/4.6.B in this Section.

** For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.

LAI



(M1)

INSERT PROPOSED NEW SPECIFICATION 3.4.7

Insert new Specification 3.4.7, Residual Heat Removal System - Hot Shutdown, as shown in the BFN Unit 2 Improved Technical Specifications.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.7
RHR SHUTDOWN COOLING SYSTEM - HOT SHUTDOWN

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A new Specification is being added requiring two RHR Shutdown Cooling subsystems to be OPERABLE in MODE 3 with reactor steam dome pressure less than the RHR low pressure permissive pressure. Appropriate ACTIONS and a Surveillance Requirement are also added. This is consistent with the BWR Standard Technical Specification, NUREG 1433 and is an additional restriction on plant operation.



(M11)

INSERT PROPOSED NEW SPECIFICATION 3.4.8

Insert new Specification 3.4.8, Residual Heat Removal System - Cold Shutdown, as shown in the BFN Unit 2 Improved Technical Specifications.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.8
RHR SHUTDOWN COOLING SYSTEM - COLD SHUTDOWN

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A new Specification is being added requiring two RHR Shutdown Cooling subsystems to be OPERABLE in MODE 4. Appropriate ACTIONS and a Surveillance Requirement are also added. This is consistent with the BWR Standard Technical Specification, NUREG 1433 and is an additional restriction on plant operation.



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.6 - RCS SPECIFIC ACTIVITY**

ADMINISTRATIVE

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 Note is added to the Required Actions for Condition A to indicate that LCO 3.0.4 is not applicable. Entry into the Applicable Modes should not be restricted since the most likely response to the condition is restoration of compliance within the allowed 48 hours. Further, since the LCO limits assure the dose due to a LOCA would be a small fraction of the 10 CFR 100 limit, operation during the allowed time frame would not represent a significant impact to the health and safety of the public. In addition, this allowance is already inherently provided by the words of Specification 4.6.B.6.a, which states that additional samples are required "during startup" when specific activity exceeds the limit. Thus, this change is a presentation preference only and is considered administrative.

- A3 Existing Specification 3.6.B.6 requires that if the Dose Equivalent I-131 cannot be restored within 48 hours, or if at any time it exceeds 26 $\mu\text{Ci/gm}$, the reactor must be shut down and all main steam lines must be isolated immediately. Proposed LCO 3.4.6, Condition B, allows the alternative of being in MODE 3 within 12 hours and Mode 4 within 36 hours under the same conditions. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In Mode 4, the LCO requirements are no longer applicable. This change is considered administrative because existing 1.0.C.1 would require that the reactor be placed in Mode 4 within 36

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.6 - RCS SPECIFIC ACTIVITY**

hours if the requirements in CTS 3.6.B.6 could not be met. This change is consistent with NUREG-1433.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 The Applicability has been changed to require the specific activity to be within limits in those conditions which represent a potential for release of significant quantities of radioactive coolant to the environment. Thus, MODE 3 with any steam line not isolated has been added. In addition, MODE 2 with any steam line not isolated has been added in lieu of MODE 2 when the reactor is critical. While this does allow the reactor to be critical with the main steam lines isolated while not requiring the LCO to be met, overall this change is considered more restrictive due to the MODE 2 subcritical and MODE 3 requirements. In addition, the ACTIONS have been modified to reflect the new Applicability, and an option for exiting the applicable MODES is provided for cases where isolation is not desired.
- M2 CTS 4.6.B.5 requires sampling reactor coolant to determine specific activity "during equilibrium power operation." Proposed SR 3.4.6.1, which contains proposed requirements for sampling reactor coolant to determine specific activity, is modified by a note that requires this Surveillance to be performed only in MODE 1. This change is slightly more restrictive because sampling will be required whenever the reactor is in MODE 1 and not just when equilibrium conditions have been established. This change is consistent with NUREG-1433.
- M3 The Surveillance Frequency has been changed from monthly to weekly (every 7 days) for consistency with NUREG-1433, Rev. 1. Since Revision 1 to the NUREG deleted the surveillance requirement to verify that reactor coolant gross specific activity is less than or equal to 100/E-bar $\mu\text{Ci/gm}$ every 7 days, the reactor coolant specific activity trending interval was decreased to 7 days from 31 days.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

- LA1 CTS 4.6.B.6 contains requirements for reactor coolant and offgas system sampling during startup, following significant power level changes, and



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.6 - RCS SPECIFIC ACTIVITY

following significant changes in offgas radiation levels. The results of any of these samples are intended to determine if RCS specific activity is exceeding specified limits. Experience has determined that the weekly sampling required by proposed SR 3.4.6.1 and requirements for monitoring main steam line and offgas radiation levels is sufficient to ensure RCS specific activity levels are not exceeded. Therefore, RCS specific activity requirements for sampling stack gas, offgas and main steam line are being relocated to plant procedures and will be controlled in accordance with the licensee controlled programs. In addition, the criteria for when specific activity has been returned to limits (for 48 hours or until a stable iodine concentration below the limit has been established with at least 3 consecutive samples being taken in all cases) has been relocated to plant procedures and will be controlled by the licensee controlled programs. The method of determining dose equivalent I-131 (i.e., quantitative measurements of specific isotopes of Iodine), as described in CTS 4.6.B.5, has also been relocated to plant procedures. These changes are consistent with NUREG-1433.

"Specific"

- L1 Proposed ACTION A allows the LCO limit to be exceeded for 48 hours provided that the specific activity does not exceed 26 $\mu\text{Ci/gm}$. CTS 3.6.B.6 allows the limit to be exceeded during a power transient and limits the time the reactor can be operated, when the LCO RCS Specific Activity limit is exceeded, to less than 5% of its yearly power operation. Generic Letter 85-19, "Reporting Requirements on Primary Coolant Iodine Spikes," states that this limit is not necessary because reactor fuel has improved significantly since this requirement was established, and that proper fuel management by licensees and existing reporting requirements for fuel failures will preclude ever approaching this limit. Removal of this limit is consistent with the BWR/4 Standard Technical Specifications, NUREG-1433, requirements.
- L2 CTS 3.6.B.6 requires the reactor to be shut down and the steam line isolation valves to be closed immediately if the iodine concentration exceeds 26 $\mu\text{Ci/gm}$. Proposed ACTION B allows 12 hours to close the isolation valves or to be in Mode 3. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam isolation valves, or to achieve the required plant conditions, in an orderly manner and without challenging plant systems. The less restrictive 12 hour Completion Time is consistent with NUREG-1433.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

(A1)

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To assure the integrity and safe operation of the reactor coolant system.

Specification

~~A. Thermal and Pressurization Limitations~~

- LCD 3.4.9 1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification

(A1) ~~A. Thermal and Pressurization Limitations~~

SR 3.4.9.1

1. During heatups and cooldowns, the following parameters shall be recorded and reactor coolant temperature determined at 15-minute intervals until 3 successive readings at each given location are within 5°F.

(L1)

(30)

(LA1)

- a. Steam Dome Pressure (Convert to upper vessel region temperature)
- b. Reactor bottom drain temperature
- c. Recirculation loops A and B
- d. Reactor vessel bottom head temperature
- e. Reactor vessel shell adjacent to shell flange

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.A. Thermal and Pressurization Limitations~~

LCO
3.4.9

2. During all operations with a critical core, other than for low-level physics tests, except when the vessel is vented, the reactor vessel shell and fluid temperatures shall be at or above the temperature of curve #3 of Figure 3.6-1.

3. During heatup by nonnuclear means, except when the vessel is vented or as indicated in 3.6.A.4, during cooldown following nuclear shutdown, or during low-level physics tests, the reactor vessel temperature shall be at or above the temperatures of curve #2 of Figure 3.6-1 until removing tension on the head stud bolts as specified in 3.6.A.5.

LCO
3.4.9

~~4.6.A. Thermal and Pressurization Limitations~~

SR 3.4.9.1

2. Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange shall be recorded at least every ~~15-30~~ (L2) minutes during inservice hydrostatic or leak testing when the vessel pressure is > 312 psig.

Note 1 to
SR 3.4.9.1

(A2)

3. Test specimens representing the reactor vessel, base weld and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-82.

SEP 13 1995

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.A. Thermal and Pressurization Limitations~~

~~4.6.A. Thermal and Pressurization Limitations~~

4. The beltline region of reactor vessel temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve #1 of Figure 3.6-1. The applicability of this curve to these tests is extended to nonnuclear heatup and ambient loss cooldown associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.

LCO 3.4.9

SR 3.4.9.1, Note 2

5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 80°F, and must remain above 80°F while under full tension.

SR 3.4.9.5 Note 2

LCO 3.4.9

AL

~~4. DELETED~~

SR 3.4.9.1, Note 1

M1

Proposed SR 3.4.9.2

SR 3.4.9.5 + Note 1
SR 3.4.9.6 + Note
SR 3.4.9.7 + Note

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

A6

LA1

Proposed frequencies for SRs 3.4.9.5, 6+7

M2



MAR 24 1993

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~3.6.A Thermal and Pressurization Limitations (Cont'd)~~

6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other. (A4)

LCD 3.4.9

7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

LCD 3.4.9

(M3)

Proposed Actions
A, B + C

~~SURVEILLANCE REQUIREMENTS~~

~~A.6.A Thermal and Pressurization Limitations (Cont'd)~~

SR 3.4.9.4 + Note 1

6. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged. (A3) (A4) (LAI)

7. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged. (LAI)

SR 3.4.9.3 + Note

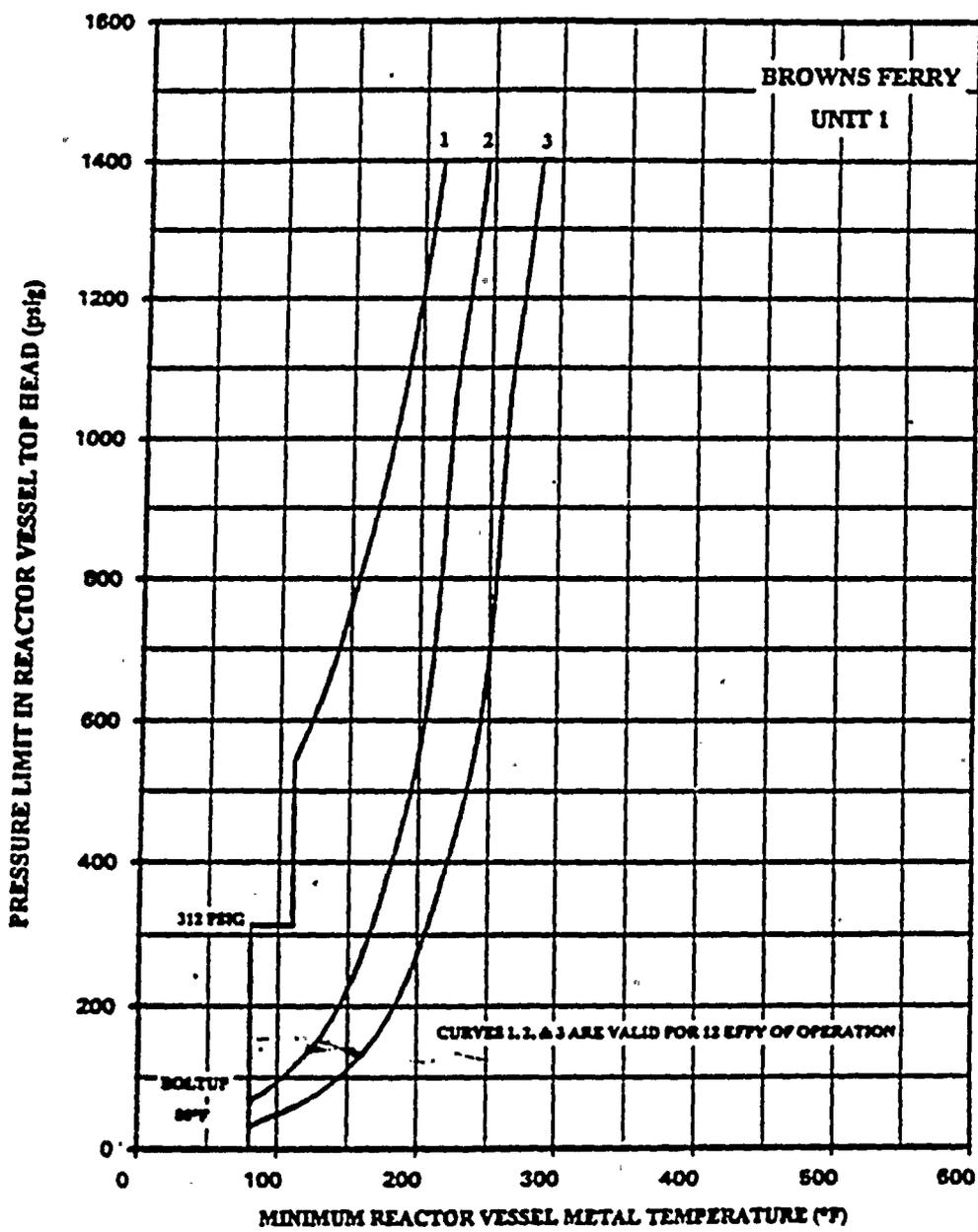
(LAI)

SEP 13 1995

3.4.9-1

Figure 3.6-1

(A1)



Curve No. 1

Minimum temperature for pressure tests such as required by Section XI. Minimum temperature of 191°F is required for test pressure of 1,100 psig.

Curve No. 2

Minimum temperature for mechanical heatup or cooldown following nuclear shutdown.

Curve No. 3

Minimum temperature for core operation (criticality) includes additional margin required for 10CFR50, Appendix G, Par. IV.A.3 which became effective July 26, 1983.

Notes

These curves include sufficient margin to provide protection against feedwater nozzle degradation. The curves allow for shifts in RT_{NDT} of the Reactor vessel belline materials, in accordance with Reg. Guide 1.89 Rev. 2, to compensate for radiation embrittlement for 12 EFPY.

3.4.9-1

(A1)

Figure 3.6-1

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

See Justification for Changes for BFN ISTS 3.4.2

See Justification for Changes for BFN ISTS 3.4.1

4.6.E. Jet Pumps

- 2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

3.6.F Recirculation Pump Operation

- 1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.
- 2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.
- 3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation

SR 3.4.9.4 Note 2

4.6.F Recirculation Pump Operation

- 1. Recirculation pump speeds shall be checked and logged at least once per day.
- 2. No additional surveillance required.

SR 3.4.9.4

(LAI)

- 3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and ~~log~~ the loop discharge temperature and dome saturation temperature.



MAY 31 1994

~~3.6/A.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.F Recirculation Pump Operation

~~3.6.F.3 (Cont'd)~~

3.4.9.4, Note 2

temperature of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

See Justification for Changes for BFN 1STS 3.4.1

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

See Justification for Changes for CTS 3.6.G/4.6.G

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.



AL

3.6.B. Coolant Chemistry

4.6.B. Coolant Chemistry

1. PRIOR TO STARTUP and at steaming rates less than 100,000 lb/hr, the following limits shall apply.

- a. Conductivity, $\mu\text{mho/cm}$ at 25°C 2.0
- b. Chloride, ppm 0.1

1. Reactor coolant shall be continuously monitored for conductivity except when there is no fuel in the reactor vessel.

- a. Whenever the continuous conductivity monitor is inoperable, a sample of reactor coolant shall be analyzed for conductivity every 4 hours except as listed below. If the reactor is in COLD SHUTDOWN CONDITION, a sample of reactor coolant shall be analyzed for conductivity every 8 hours.
- b. Once a week the continuous monitor shall be checked with an in-line flow cell. This in-line conductivity calibration shall be performed every 24 hours whenever the reactor coolant conductivity is $>1.0 \mu\text{mho/cm}$ at 25°C.

2. At steaming rates greater than 100,000 lb/hr, the following limits shall apply.

- a. Conductivity, $\mu\text{mho/cm}$ at 25°C 1.0
- b. Chloride, ppm 0.2

2. During startup prior to pressurizing the reactor above atmospheric pressure, measurements of reactor water quality shall be performed to show conformance with 3.6.B.1 of limiting conditions.

(R1) →

(A1)

3.6.B. Coolant Chemistry

3. At steaming rates greater than 100,000 lb/hr, the reactor water quality may exceed Specification 3.6.B.2 only for the time limits specified below. Exceeding these time limits or the following maximum quality limits shall be cause for placing the reactor in the **COLD SHUTDOWN CONDITION**.

a. Conductivity time above 1 μ ho/cm at 25°C - 2 weeks/year.
Maximum Limit 10 μ ho/cm at 25°C

b. Chloride concentration time above 0.2 ppm - 2 weeks/year.
Maximum Limit - 0.5 ppm.

c. The reactor shall be placed in the **SHUTDOWN CONDITION** if pH <5.6 or >8.6 for a 24-hour period.

(R1)

4.6.B. Coolant Chemistry

3. Whenever the reactor is operating (including **HOT STANDBY CONDITION**) measurements of reactor water quality shall be performed according to the following schedule:

a. Chloride ion content and pH shall be measured at least once every 96 hours.

b. Chloride ion content shall be measured at least every 8 hours whenever reactor conductivity is >1.0 μ ho/cm at 25°C.

c. A sample of reactor coolant shall be measured for pH at least once every 8 hours whenever the reactor coolant conductivity is >1.0 μ ho/cm at 25°C.



JUN 28 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.B. Coolant Chemistry

4. When the reactor is not pressurized with fuel in the reactor vessel, except during the STARTUP CONDITION, the reactor water shall be maintained within the following limits.

- a. Conductivity - 10 μ ho/cm at 25°C
- b. Chloride - 0.5 ppm
- c. pH shall be between 5.3 and 8.6.

5. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated immediately. The reactor shall be brought to the COLD SHUTDOWN CONDITION as rapidly as cooldown rate permits.

6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 μ Ci/gm of dose equivalent I-131.

See Justification for changes to BFN ISTS 3.4.6 in this section

4.6.B. Coolant Chemistry

4. Whenever the reactor is not pressurized with fuel in the reactor vessel, a sample of the reactor coolant shall be analyzed at least every 96 hours for conductivity, chloride ion content and pH.

5. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least I-131, I-132, I-133, and I-134 shall be performed monthly on a coolant liquid sample.

6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:

(AI)

3.6.B. Coolant Chemistry

3.6.B.6 (Cont'd)

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 $\mu\text{Ci/gm}$ whenever the reactor is critical. The reactor shall not be operated more than 5% of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 $\mu\text{Ci/gm}$, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

See Justification for Changes for BFN ISTS 3.4.6 in this section

(R1)

7. When there is no fuel in the reactor vessel, technical specification reactor coolant chemistry limits do not apply.

(R1)

7. When there is no fuel in the reactor vessel, sampling of reactor coolant chemistry at technical specification frequency is not required.

** For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.

4.6.B. Coolant Chemistry

4.6.B.6 (Cont'd)

- a. During the STARTUP CONDITION
- b. Following a significant power change**
- c. Following an increase in the equilibrium off-gas level exceeding 10,000 $\mu\text{Ci/sec}$ (at the steam jet air ejector) within a 48-hour period.
- d. Whenever the equilibrium iodine limit specified in 3.6.B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 $\mu\text{Ci/gm}$) is established. However, at least 3 consecutive samples shall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration.



MAY 31 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

temperature of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

See Justification for Changes to BFN 1STS 3.4.1, Recirculation Loops operating, in this section

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

(R1)

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.



MAY 3 1 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.G Structural Integrity

3.6.G.1 (Cont'd)

(R1) →

- b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.

4.6.G. Structural Integrity

(R1) →

- 3. For Unit 1 an augmented inservice surveillance program shall be performed to monitor potential corrosive effects of chloride residue released during the March 22, 1975 fire. The augmented inservice surveillance program is specified as follows:
 - a. Browns Ferry Mechanical Maintenance Instruction 53, dated September 22, 1975, paragraph 4, defines the liquid penetrant examinations required during the first, second, third and fourth refueling outages following the fire restoration.
 - b. Browns Ferry Mechanical Maintenance Instruction 46, dated July 18, 1975, Appendix B, defines the liquid penetrant examinations required during the sixth refueling outage following the fire restoration.

JUL 05 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(AI)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.H. Snubbers

During all modes of operation, all snubbers shall be OPERABLE except as noted in 3.6.H.1. All safety-related snubbers are listed in Plant Surveillance Instructions.

1. With one or more snubber(s) inoperable on a system that is required to be OPERABLE in the current plant condition, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system inoperable and follow the appropriate Limiting Condition statement for that system.

(LAI)

4.6.H. Snubbers

Each safety-related snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 3.6.H/4.6.H. These snubbers are listed in Plant Surveillance Instructions.

1. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

2. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.6.H-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.6.H-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment No. 210.



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4.6.H. Snubbers

3. Visual Inspection
Acceptance Criteria

Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.6.H.5. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the LIMITING CONDITIONS FOR OPERATION shall be met.

(LAI)



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

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~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

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4.6.H Snubbers

4.6.H.3 (Cont'd)

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Additionally, snubbers attached to sections of safety-related systems that have experienced unexpected potentially damaging transients since the last inspection period shall be evaluated for the possibility of concealed damage and functionally tested, if applicable, to confirm OPERABILITY. Snubbers which have been made inoperable as the result of unexpected transients, isolated damage, or other random events, when the provisions of 4.6.H.7 and 4.6.H.8 have been met and any other appropriate corrective action implemented, shall not be counted in determining the next visual inspection interval.

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

JUL 05 1994

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

4.6.H. Snubbers

4. FUNCTIONAL TEST Schedule, Lot Size, and Composition

During each refueling outage, a representative sample of 10% of the total of each type of safety-related snubbers in use in the plant shall be functionally tested either in place or in a bench test.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of snubbers within the types. The representative sample should be weighed to include more snubbers from severe service areas such as near heavy equipment.

The stroke setting and the security of fasteners for attachment of the snubbers to the component and to the snubber anchorage shall be verified on snubbers selected for FUNCTIONAL TESTS.

(2A1) →



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

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~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

4.6.H. Snubbers

5. FUNCTIONAL TEST Acceptance Criteria

The snubber FUNCTIONAL TEST shall verify that:

(LA1)

- a. Activation (restraining action) is achieved in both tension and compression within the specified range, except that inertia dependent, acceleration limiting mechanical snubbers may be tested to verify only that activation takes place in both directions of travel.
- b. Snubber bleed, or release where required, is present in both compression and tension within the specified range.
- c. For mechanical snubbers, the force required to initiate or maintain motion of the snubber is not great enough to overstress the attached piping or component during thermal movement, or to indicate impending failure of the snubber.
- d. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.



JUL 05 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

4.6.H. Snubbers

4.6.H.5 (Cont'd)

e. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

6. FUNCTIONAL TEST Failure Analysis and Additional Test/Lots

(LAI)

An engineering evaluation shall be made of each failure to meet the FUNCTIONAL TEST acceptance criteria to determine the cause of the failure. The result of this analysis shall be used, if applicable, in selecting snubbers to be tested in the subsequent lot in an effort to determine the OPERABILITY of other snubbers which may be subject to the same failure mode. Selection of snubbers for future testing may also be based on the failure analysis. For each snubber that does not meet the FUNCTIONAL TEST acceptance criteria, an additional lot equal to 10 percent of the remainder of that type of snubbers shall be functionally tested. Testing shall continue until no additional inoperable snubbers are found within subsequent lots or all snubbers of the original FUNCTIONAL TEST type have been tested or all suspect snubbers identified by the failure analysis have been tested, as applicable.



JAN 19 1989

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4.6.H. Snubbers

4.6.H.6 (Cont'd)

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency, all snubbers of the same design subject to the same defect shall be functionally tested.

This testing requirement shall be independent of the requirements stated above for snubbers not meeting the FUNCTIONAL TEST acceptance criteria.

The discovery of loose or missing attachment fasteners will be evaluated to determine whether the cause may be localized or generic. The result of the evaluation will be used to select other suspect snubbers for verifying the attachment fasteners, as applicable.

7. FUNCTIONAL TEST Failure - Attached Component Analysis

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are restrained by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components restrained by the

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AMENDMENT NO. 163



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

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~~LIMITING CONDITIONS FOR OPERATION~~~~SURVEILLANCE REQUIREMENTS~~

4.6.H. Snubbers

4.6.H.7 (Cont'd)

snubber(s) were adversely affected by the inoperability of the snubber(s), and in order to ensure that the restrained component remains capable of meeting the designed service.

8. Functional Testing Of Repaired and Spare Snubbers

Snubbers which fail the visual inspection or the FUNCTIONAL TEST acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the FUNCTIONAL TEST results shall meet the FUNCTIONAL TEST criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the FUNCTIONAL TEST must have been performed within 12 months before being installed in the unit.

9. Exemption from Visual Inspection or FUNCTIONAL TESTS

Permanent or other exemptions from visual inspections and/or functional testing for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and if applicable snubber life destructive testing was performed to qualify snubber OPERABILITY for the applicable design conditions at either the

(2A1)



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4.6.H. Snubbers

4.6.H.9 (Cont'd)

completion of their fabrication or at a subsequent date. Snubbers so exempted shall continue to be listed in the plant instructions with footnotes indicating the extent of the exemptions.

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10. Snubber Service Life Program

The service life of snubbers may be extended based on an evaluation of the records of FUNCTIONAL TESTS, maintenance history, and environmental conditions to which the snubbers have been exposed.



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Table 4.6.H-1

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SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS		
	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or more	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.



JUL 05 1994

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Table 4.6.H-1 (Continued)

SNUBBER VISUAL INSPECTION INTERVAL

- Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.
- Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C, but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.
- Note 6: The provisions of Specification 1.0.II are applicable for all inspection intervals up to and including 48 months.



Section 3.4, Reactor Coolant System (RCS) Bases

The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of the proposed Browns Ferry Unit 2 Technical Specification Section 3.4, consistent with the BWR Standard Technical Specification, NUREG 1433. The revised Bases are as shown in the proposed Browns Ferry Unit 2 Technical Specification Bases.

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



AI

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To assure the integrity and safe operation of the reactor coolant system.

Specification

~~A. Thermal and Pressurization Limitations~~

- LC0 3.4.9 1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification

AI

~~A. Thermal and Pressurization Limitations~~

SR 3.4.9.1

1. ^{Note 1 to SR 3.4.9.1} During heatups and cooldowns, the following parameters shall be recorded and reactor coolant temperature determined at 15-minute intervals until 3 successive readings at each given location are within 5°F.

L1 30

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- a. Steam Dome Pressure (Convert to upper vessel region temperature)
- b. Reactor bottom drain temperature
- c. Recirculation loops A and B
- d. Reactor vessel bottom head temperature
- e. Reactor vessel shell adjacent to shell flange



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.A. Thermal and Pressurization Limitations~~

(A1)

~~4.6.A. Thermal and Pressurization Limitations~~

SR 3.4.9.1

LCO
3.4.9
2. During all operations with a critical core, other than for low-level physics tests, except when the vessel is vented, the reactor vessel shell and fluid temperatures shall be at or above the temperature of curve #3 of Figure 3.6-1.

2. Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange shall be recorded at least every 15 30' (L2) minutes during inservice hydrostatic or leak testing when the vessel pressure is > 312 psig.

Note 1 to SR 3.4.9.1

LCO
3.4.9
3. During heatup by nonnuclear means, except when the vessel is vented or as indicated in 3.6.A.4, during cooldown following nuclear shutdown, or during low-level physics tests, the reactor vessel temperature shall be at or above the temperatures of curve #2 of Figure 3.6-1 until removing tension on the head stud bolts as specified in 3.6.A.5.

(A2)

3. Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-82.



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~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.A. Thermal and Pressurization Limitations (Cont'd)~~

~~4.6.A. Thermal and Pressurization Limitations (Cont'd)~~

4. The beltline region of reactor vessel temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve #1 of Figure 3.6-1. The applicability of this curve to these tests is extended to nonnuclear heatup and ambient loss cooldown associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.

LCO 3.4.9

SR 3.4.9.1, Note 2

SR 3.4.9.5 Note 2

LCO 3.4.9

(A1)

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SR 3.4.9.1, Note 1.

(M1)

Proposed SR 3.4.9.2

SR 3.4.9.5 + Note 1
SR 3.4.9.6 + Note
SR 3.4.9.7 + Note

5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 82°F, and must remain above 82°F while under full tension.

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

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(AG)

Proposed frequencies for SRs 3.4.9.5, 6 + 7

(M2)



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~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~3.6.A Thermal and Pressurization Limitations (Cont'd)~~

LCO
3.4.9
6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other. -

LCO
3.4.9
7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

M3
Proposed ACTIONS
A, B + C

~~SURVEILLANCE REQUIREMENTS~~

~~4.6.A Thermal and Pressurization Limitations (Cont'd)~~

SR 34.9.4 + Note 1

A1
A3
A4
LA1
6. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

SR 3.4.9.3

Note

7. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged. LA1



LIMITING CONDITIONS FOR OPERATION

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SURVEILLANCE REQUIREMENTS

See Justification for Changes for BFN ISTS 3.4.2

See Justification for Changes for BFN ISTS 3.4.1

4.6.E. Jet Pumps

- 2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

3.6.F. Recirculation Pump Operation

- 1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.
- 2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.
- 3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval, restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature of the reactor

SR 3.4.9.4 Note 2

4.6.F. Recirculation Pump Operation

- 1. Recirculation pump speeds shall be checked and logged at least once per day.
- 2. No additional surveillance required.

SR 3.4.9.4

(LAI)

- 3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.

MAR 18 1993

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

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~~SURVEILLANCE REQUIREMENTS~~

3.6.F Recirculation Pump Operation

~~3.6.F.3 (Cont'd)~~

3.4.9.4, Note 2
vessel water as determined
by dome pressure. The
total elapsed time in natural
circulation and one pump
operation must be no greater
than 24 hours.

- 4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

See Justification for Change
for BAN ISTS 3.4.1

3.6.G Structural Integrity

- 1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a COLD SHUTDOWN CONDITION or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

4.6.G Structural Integrity

- 1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
- 2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

See Justification for Changes
for CTS 3.6.G/4.6.G

3.6/4.6-13

BFN
Unit 2

AMENDMENT NO. 206

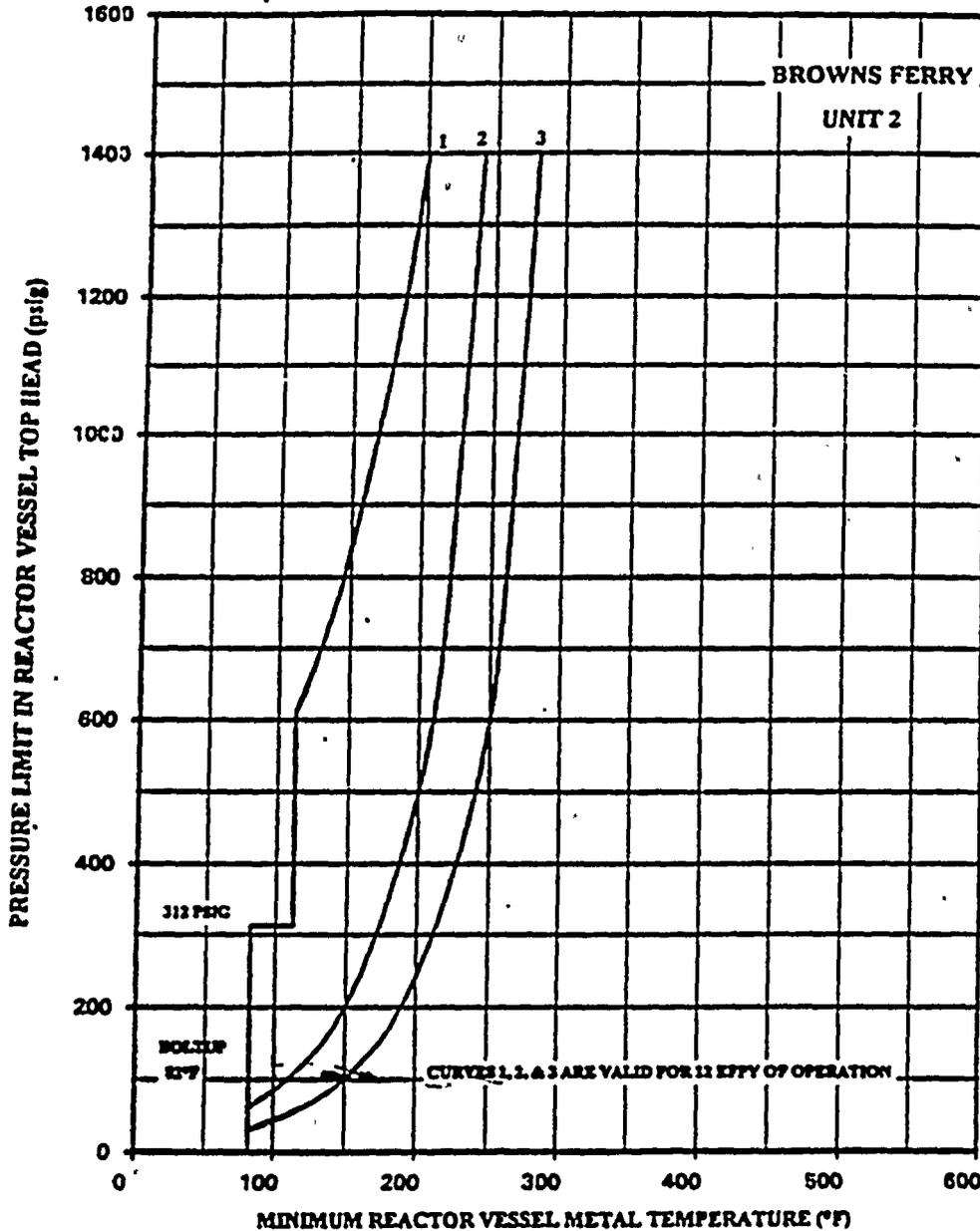


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3.4.9-1

Figure 3.6-1



Curve No. 1

Minimum temperature for pressure tests such as required by Section XI. Minimum temperature of 178°F is required for test pressure of 1,100 psig.

Curve No. 2

Minimum temperature for mechanical heatup or cooldown following nuclear shutdown.

Curve No. 3

Minimum temperature for core operation (criticality) includes additional margin required for 10CFR50, Appendix G, Par. IV.A.3 which became effective July 26, 1983.

Notes

These curves include sufficient margin to provide protection against feedwater nozzle degradation. The curves allow for shifts in RT_{NDT} of the Reactor vessel belline materials, in accordance with Reg. Guide 1.99 Rev. 2, to compensate for radiation embrittlement for 12 EFFY.

3.4.9-1

(A1)

Figure 3.6-1



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3.6.B. Coolant Chemistry

1. PRIOR TO STARTUP and at steaming rates less than 100,000 lb/hr, the following limits shall apply.

a. Conductivity, $\mu\text{mho/cm}$ at 25°C 2.0

b. Chloride, ppm 0.1

2. At steaming rates greater than 100,000 lb/hr, the following limits shall apply.

a. Conductivity, $\mu\text{mho/cm}$ at 25°C 1.0

b. Chloride, ppm 0.2

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4.6.B. Coolant Chemistry

1. Reactor coolant shall be continuously monitored for conductivity except when there is no fuel in the reactor vessel.

a. Whenever the continuous conductivity monitor is inoperable, a sample of reactor coolant shall be analyzed for conductivity every 4 hours except as listed below. If the reactor is in COLD SHUTDOWN CONDITION, a sample of reactor coolant shall be analyzed for conductivity every 8 hours.

b. Once a week the continuous monitor shall be checked with an in-line flow cell. This in-line conductivity calibration shall be performed every 24 hours whenever the reactor coolant conductivity is $>1.0 \mu\text{mho/cm}$ at 25°C.

2. During startup prior to pressurizing the reactor above atmospheric pressure, measurements of reactor water quality shall be performed to show conformance with 3.6.B.1 of limiting conditions.

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3.6.B. Coolant Chemistry

3. At steaming rates greater than 100,000 lb/hr, the reactor water quality may exceed Specification 3.6.B.2 only for the time limits specified below. Exceeding these time limits or the following maximum quality limits shall be cause for placing the reactor in the **COLD SHUTDOWN CONDITION**.

a. Conductivity
time above
1 $\mu\text{mho/cm}$ at 25°C -
2 weeks/year.
Maximum Limit
10 $\mu\text{mho/cm}$ at 25°C

b. Chloride
concentration time
above 0.2 ppm -
2 weeks/year.
Maximum Limit -
0.5 ppm.

c. The reactor shall be placed in the **SHUTDOWN CONDITION** if pH <5.6 or >8.6 for a 24-hour period.

4.6.B. Coolant Chemistry

3. Whenever the reactor is operating (including **HOT STANDBY CONDITION**) measurements of reactor water quality shall be performed according to the following schedule:

a. Chloride ion content and pH shall be measured at least once every 96 hours.

b. Chloride ion content shall be measured at least every 8 hours whenever reactor conductivity is >1.0 $\mu\text{mho/cm}$ at 25°C.

c. A sample of reactor coolant shall be measured for pH at least once every 8 hours whenever the reactor coolant conductivity is >1.0 $\mu\text{mho/cm}$ at 25°C.

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3.6.B. Coolant Chemistry

4.6.B. Coolant Chemistry

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- 4. When the reactor is not pressurized with fuel in the reactor vessel, except during the STARTUP CONDITION, the reactor water shall be maintained within the following limits.
 - a. Conductivity - 10 μ mho/cm at 25°C
 - b. Chloride - 0.5 ppm
 - c. pH shall be between 5.3 and 8.6.
- 5. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated immediately. The reactor shall be brought to the COLD SHUTDOWN CONDITION as rapidly as cooldown rate permits.
- 6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 μ Ci/gm of dose equivalent I-131.

- 4. Whenever the reactor is not pressurized with fuel in the reactor vessel, a sample of the reactor coolant shall be analyzed at least every 96 hours for conductivity, chloride ion content and pH.
- 5. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least I-131, I-132, I-133, and I-134 shall be performed monthly on a coolant liquid sample.
- 6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:

SEE JUSTIFICATION FOR CHANGES TO BFN ISTS 3.4.6 IN THIS SECTION



JUN 28 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

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~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.B. Coolant Chemistry

3.6.B.6 (Cont'd)

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 $\mu\text{Ci/gm}$ whenever the reactor is critical. The reactor shall not be operated more than 5% of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 $\mu\text{Ci/gm}$, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

4.6.B. Coolant Chemistry

4.6.B.6 (Cont'd)

- a. During the STARTUP CONDITION
- b. Following a significant power change**
- c. Following an increase in the equilibrium off-gas level exceeding 10,000 $\mu\text{Ci/sec}$ (at the steam jet air ejector) within a 48-hour period.
- d. Whenever the equilibrium iodine limit specified in 3.6.B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 $\mu\text{Ci/gm}$) is established. However, at least 3 consecutive samples shall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration.

SEE JUSTIFICATION FOR CHANGES FOR BFN 1STS 3.4.6 IN THIS SECTION

RI

7. When there is no fuel in the reactor vessel, technical specification reactor coolant chemistry limits do not apply.

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7. When there is no fuel in the reactor vessel, sampling of reactor coolant chemistry at technical specification frequency is not required.

** For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.

MAR 18 1993

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

AD

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

See Justification for Changes to BFN 15TS 3.4.1, Recirculation Loops Operating, in this Section.

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a COLD SHUTDOWN CONDITION or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

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4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.



MAR 18 1993

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

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~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.G Structural Integrity

3.6.G.1 (Cont'd)

(R1)

- b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.



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~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.H. Snubbers

During all modes of operation, all snubbers shall be OPERABLE except as noted in 3.6.H.1. All safety-related snubbers are listed in Plant Surveillance Instructions.

1. With one or more snubber(s) inoperable on a system that is required to be OPERABLE in the current plant condition, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system inoperable and follow the appropriate Limiting Condition statement for that system.

LAI

4.6.H. Snubbers

Each safety-related snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 3.6.H/4.6.H. These snubbers are listed in Plant Surveillance Instructions.

1. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

2. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.6.H-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.6.H-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment No. 225.

JUL 05 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

4.6.H. Snubbers

3. Visual Inspection Acceptance Criteria

Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY; (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.6.H.5. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the LIMITING CONDITIONS FOR OPERATION shall be met.

(LAI) →



JUL 05 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(AI)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

4.6.H Snubbers

4.6.H.3 (Cont'd)

(LAI)

~~Additionally, snubbers attached to sections of safety-related systems that have experienced unexpected potentially damaging transients since the last inspection period shall be evaluated for the possibility of concealed damage and functionally tested, if applicable, to confirm OPERABILITY. Snubbers which have been made inoperable as the result of unexpected transients, isolated damage, or other random events, when the provisions of 4.6.H.7 and 4.6.H.8 have been met and any other appropriate corrective action implemented, shall not be counted in determining the next visual inspection interval.~~



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JUL 05 1994

4.6.H. Snubbers

4. FUNCTIONAL TEST Schedule, Lot Size, and Composition

During each refueling outage, a representative sample of 10% of the total of each type of safety-related snubbers in use in the plant shall be functionally tested either in place or in a bench test.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of snubbers within the types. The representative sample should be weighed to include more snubbers from severe service areas such as near heavy equipment.

The stroke setting and the security of fasteners for attachment of the snubbers to the component and to the snubber anchorage shall be verified on snubbers selected for FUNCTIONAL TESTS.

LAI





(A1)

JAN 19 1989

4.6.H. Snubbers

5. FUNCTIONAL TEST Acceptance Criteria

The snubber FUNCTIONAL TEST shall verify that:

(LAI) →

- a. Activation (restraining action) is achieved in both tension and compression within the specified range, except that inertia dependent, acceleration limiting mechanical snubbers may be tested to verify only that activation takes place in both directions of travel.
- b. Snubber bleed, or release where required, is present in both compression and tension within the specified range.
- c. For mechanical snubbers, the force required to initiate or maintain motion of the snubber is not great enough to overstress the attached piping or component during thermal movement, or to indicate impending failure of the snubber.
- d. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.



JUL 05 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

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~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

4.6.H. Snubbers

4.6.H.5 (Cont'd)

e. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

6. FUNCTIONAL TEST Failure Analysis and Additional Test Lots

LAI

An engineering evaluation shall be made of each failure to meet the FUNCTIONAL TEST acceptance criteria to determine the cause of the failure. The result of this analysis shall be used, if applicable, in selecting snubbers to be tested in the subsequent lot in an effort to determine the OPERABILITY of other snubbers which may be subject to the same failure mode. Selection of snubbers for future testing may also be based on the failure analysis. For each snubber that does not meet the FUNCTIONAL TEST acceptance criteria, an additional lot equal to 10 percent of the remainder of that type of snubbers shall be functionally tested. Testing shall continue until no additional inoperable snubbers are found within subsequent lots or all snubbers of the original FUNCTIONAL TEST type have been tested or all suspect snubbers identified by the failure analysis have been tested, as applicable.

JAN 19 1989

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4.6.H. Snubbers

4.6.H.6 (Cont'd)

LAI

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency, all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the FUNCTIONAL TEST acceptance criteria.

The discovery of loose or missing attachment fasteners will be evaluated to determine whether the cause may be localized or generic. The result of the evaluation will be used to select other suspect snubbers for verifying the attachment fasteners, as applicable.

7. FUNCTIONAL TEST Failure - Attached Component Analysis

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are restrained by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components restrained by the

AI

4.6.H. Snubbers

4.6.H.7 (Cont'd)

snubber(s) were adversely affected by the inoperability of the snubber(s), and in order to ensure that the restrained component remains capable of meeting the designed service.

LAI

8. Functional Testing Of Repaired and Spare Snubbers

Snubbers which fail the visual inspection or the FUNCTIONAL TEST acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the FUNCTIONAL TEST results shall meet the FUNCTIONAL TEST criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the FUNCTIONAL TEST must have been performed within 12 months before being installed in the unit.

9. Exemption from Visual Inspection or FUNCTIONAL TESTS

Permanent or other exemptions from visual inspections and/or functional testing for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and if applicable snubber life destructive testing was performed to qualify snubber OPERABILITY for the applicable design conditions at either the



(AI)

JAN 19 1989

4.6.H. Snubbers

4.6.H.9 (Cont'd)

completion of their fabrication or at a subsequent date. Snubbers so exempted shall continue to be listed in the plant instructions with footnotes indicating the extent of the exemptions.

(LAI) →

10. Snubber Service Life Program

The service life of snubbers may be extended based on an evaluation of the records of FUNCTIONAL TESTS, maintenance history, and environmental conditions to which the snubbers have been exposed.



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LA1

Table 4.6.H-1
SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS		
	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or more	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.



JUL 05 1994

LAI

Table 4.6.H-1 (Continued)

SNUBBER VISUAL INSPECTION INTERVAL

- Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.
- Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C, but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.
- Note 6: The provisions of Specification 1.0.LL are applicable for all inspection intervals up to and including 48 months.

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To assure the integrity and safe operation of the reactor coolant system.

Specification

~~A. Thermal and Pressurization Limitations~~

LCO 3.4.9

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification

~~A. Thermal and Pressurization Limitations~~

(A1)

SR 3.4.9.1

Note: 1. to SR 3.4.9.1

(L1)

(30)

During heatups and cooldowns, the following parameters shall be recorded and reactor coolant temperature determined at 15-minute intervals until 3 successive readings at each given location are within 5°F.

(LA1)

- a. Steam Dome Pressure (Convert to upper vessel region temperature)
- b. Reactor bottom drain temperature
- c. Recirculation loops A and B
- d. Reactor vessel bottom head temperature
- e. Reactor vessel shell adjacent to shell flange

(A1)

~~3.6.A. Thermal and Pressurization Limitations~~

LCO
3.4.9

2. During all operations with a critical core, other than for low-level physics tests, except when the vessel is vented, the reactor vessel shell and fluid temperatures shall be at or above the temperature of curve #3 of Figure 3.6-1.

LCO
3.4.9

3. During heatup by nonnuclear means, except when the vessel is vented or as indicated in 3.6.A.4, during cooldown following nuclear shutdown, or during low-level physics tests, the reactor vessel temperature shall be at or above the temperatures of curve #2 of Figure 3.6-1 until removing tension on the head stud bolts as specified in 3.6.A.5.

~~4.6.A. Thermal and Pressurization Limitations~~

SR 3.4.9.1

2. Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange shall be recorded at least every ~~15-30~~ L2 minutes during inservice hydrostatic or leak testing when the vessel pressure is > 312 psig.

Note 1 to SR 3.4.9.1

(A2)

3. Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-82.



(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.A. Thermal and Pressurization Limitations~~

~~A.6.A. Thermal and Pressurization Limitations~~

4. The beltline region of reactor vessel temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve #1 of Figure 3.6-1. The applicability of this curve to these tests is extended to nonnuclear heatup and ambient loss cooldown associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.

LCO 3.4.9

SR 3.4.9.1 Note 2

5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 70°F, and must remain above 70°F while under full tension.

SR 3.4.9.1 Note 2

LCO 3.4.9

~~A. DELETED~~

SR 3.4.9.1, Note 1

(M1)

Proposed SR 3.4.9.2

SR 3.4.9.5 + Note 1
 SR 3.4.9.6 + Note
 SR 3.4.9.7 + Note

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

(A6)

(LA1)

Proposed Frequencies for SRs 3.4.9.5, 6 & 7

(M2)



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~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

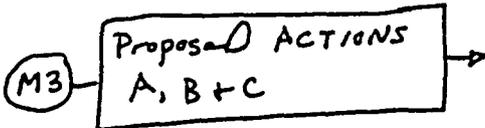
~~3.6.A Thermal and Pressurization Limitations (Cont'd)~~ (A1)

LCO
3.4.9

6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

LCO
3.4.9

7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and bottom head drain are within 145°F.



~~SURVEILLANCE REQUIREMENTS~~

~~4.6.A Thermal and Pressurization Limitations (Cont'd)~~

SR3.4.9.4 + Note 1

6. Prior to and during STARTUP of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged. (A4) (LA1)

SR3.4.9.37.
+
Note

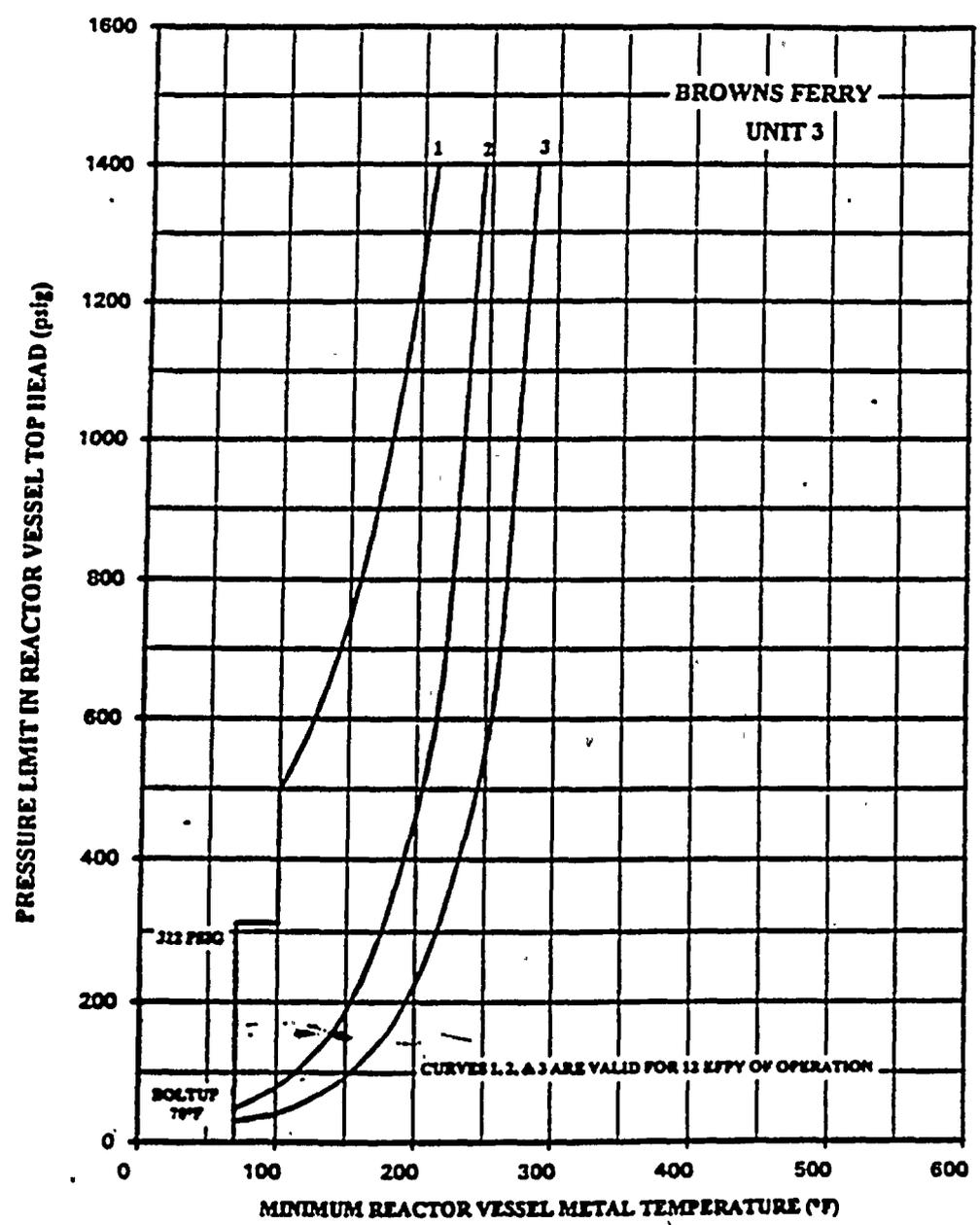
Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

(LA1)



SEP 13 1995

(A1) Figure 3.4.9-1
~~Figure 3.6-1~~



Curve No. 1
Minimum temperature for pressure tests such as required by Section XI. Minimum temperature of 189°F is required for test pressure of 1,100 psig.

Curve No. 2
Minimum temperature for mechanical heatup or cooldown following nuclear shutdown.

Curve No. 3
Minimum temperature for core operation (criticality) includes additional margin required for 10CFR50, Appendix G, Par. IV.A.3 which became effective July 26, 1983.

Notes
These curves include sufficient margin to provide protection against feedwater nozzle degradation. The curves allow for shifts in RT_{NDT} of the Reactor vessel belline materials, in accordance with Reg. Guide 1.99 Rev. 2, to compensate for radiation embrittlement for 12 EFFY.

Figure 3.4.9(A1)
~~Figure 3.6-1~~



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(AI)

See Justification for changes for BFN ISTS 3.4.2

See Justification for changes for BFN ISTS 3.4.1

4.6.E. Jet Pumps

- 2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

3.6.F Recirculation Pump Operation

- 1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.
- 2. Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.
- 3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature

4.6.F Recirculation Pump Operation

- 1. Recirculation pump speeds shall be checked and logged at least once per day.
- 2. No additional surveillance required.

SR 3.4.9.4

- 3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.

(LAL)

SR 4.9.4 of 2

(A1)

3.6.F Recirculation Pump Operation

~~3.6.F.3 (Cont'd)~~

3.4.9.4, Note 2

of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

- 4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

See Justification for Changes for BFN 1STS 3.4.1

3.6.G Structural Integrity

- 1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

4.6.G Structural Integrity

- 1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
- 2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

See Justification for Changes for CTS 3.6.G/4.6.G

(A1)

JUN 28 1994

3.6.B. Coolant Chemistry

4.6.B. Coolant Chemistry

1. PRIOR TO STARTUP and at steaming rates less than 100,000 lb/hr, the following limits shall apply.
 - a. Conductivity, $\mu\text{mho/cm}$ at 25°C 2.0
 - b. Chloride, ppm 0.1

2. At steaming rates greater than 100,000 lb/hr, the following limits shall apply.
 - a. Conductivity, $\mu\text{mho/cm}$ at 25°C 1.0
 - b. Chloride, ppm 0.2

1. Reactor coolant shall be continuously monitored for conductivity except when there is no fuel in the reactor vessel.
 - a. Whenever the continuous conductivity monitor is inoperable, a sample of reactor coolant shall be analyzed for conductivity every 4 hours except as listed below. If the reactor is in COLD SHUTDOWN CONDITION, a sample of reactor coolant shall be analyzed for conductivity every 8 hours.
 - b. Once a week the continuous monitor shall be checked with an in-line flow cell. This in-line conductivity calibration shall be performed every 24 hours whenever the reactor coolant conductivity is $>1.0 \mu\text{mho/cm}$ at 25°C.

2. During startup prior to pressurizing the reactor above atmospheric pressure, measurements of reactor water quality shall be performed to show conformance with 3.6.B.1 of limiting conditions.

(R1)



DEC 07 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

AD

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.B. Coolant Chemistry

3. At steaming rates greater than 100,000 lb/hr, the reactor water quality may exceed Specification 3.6.B.2 only for the time limits specified below. Exceeding these time limits or the following maximum quality limits shall be cause for placing the reactor in the COLD SHUTDOWN CONDITION.
 - a. Conductivity time above 1 $\mu\text{mho/cm}$ at 25°C - 2 weeks/year.
Maximum Limit 10 $\mu\text{mho/cm}$ at 25°C
 - b. Chloride concentration time above 0.2 ppm - 2 weeks/year.
Maximum Limit - 0.5 ppm.
 - c. The reactor shall be placed in the SHUTDOWN CONDITION if pH <5.6 or >8.6 for a 24-hour period.

4.6.B. Coolant Chemistry

3. Whenever the reactor is operating (including HOT STANDBY CONDITION) measurements of reactor water quality shall be performed according to the following schedule:
 - a. Chloride ion content and pH shall be measured at least once every 96 hours.
 - b. Chloride ion content shall be measured at least every 8 hours whenever reactor conductivity is >1.0 $\mu\text{mho/cm}$ at 25°C.
 - c. A sample of reactor coolant shall be measured for pH at least once every 8 hours whenever the reactor coolant conductivity is >1.0 $\mu\text{mho/cm}$ at 25°C.

RI



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3.6.B. Coolant Chemistry

4. When the reactor is not pressurized with fuel in the reactor vessel, except during the STARTUP CONDITION, the reactor water shall be maintained within the following limits.

R1

- a. Conductivity - 10 $\mu\text{mho/cm}$ at 25°C
- b. Chloride - 0.5 ppm
- c. pH shall be between 5.3 and 8.6.

5. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated immediately. The reactor shall be brought to the COLD SHUTDOWN CONDITION as rapidly as cooldown rate permits.

6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 $\mu\text{Ci/gm}$ of dose equivalent I-131.

See Justification for Changes to BFN ISTS 3.4.6 In this Section

4.6.B. Coolant Chemistry

4. Whenever the reactor is not pressurized with fuel in the reactor vessel, a sample of the reactor coolant shall be analyzed at least every 96 hours for conductivity, chloride ion content and pH.

5. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least I-131, I-132, I-133, and I-134 shall be performed monthly on a coolant liquid sample.

6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:



AI

3.6.B. Coolant Chemistry

3.6.B.6 (Cont'd)

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 $\mu\text{Ci/gm}$ whenever the reactor is critical. The reactor shall not be operated more than 5% of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 $\mu\text{Ci/gm}$, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

See Justification for Changes
BFN 15 TS 3.4.6 in this
Section

RI

7. When there is no fuel in the reactor vessel, technical specification reactor coolant chemistry limits do not apply.

4.6.B. Coolant Chemistry

4.6.B.6 (Cont'd)

- a. During the STARTUP CONDITION
- b. Following a significant power change**
- c. Following an increase in the equilibrium off-gas level exceeding 10,000 $\mu\text{Ci/sec}$ (at the steam jet air ejector) within a 48-hour period.
- d. Whenever the equilibrium iodine limit specified in 3.6.B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 $\mu\text{Ci/gm}$) is established. However, at least 3 consecutive samples shall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration.

RI

7. When there is no fuel in the reactor vessel, sampling of reactor coolant chemistry at technical specification frequency is not required.

** For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.



MAY 31 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(AI)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

see justification for Changes to BFN ISTS 3.4.1, Recirculation Loops operating, in this section

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.

(RI)

- a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.G Structural Integrity

3.6.G.1 (Cont'd)

(R1)

- b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.

4.6.G Structural Integrity



JUL 05 1994

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3.6.H. Snubbers

During all modes of operation, all snubbers shall be OPERABLE except as noted in 3.6.H.1. All safety-related snubbers are listed in Plant Surveillance Instructions.

1. With one or more snubber(s) inoperable on a system that is required to be OPERABLE in the current plant condition, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system inoperable and follow the appropriate Limiting Condition statement for that system.

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4.6.H. Snubbers

Each safety-related snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 3.6.H/4.6.H. These snubbers are listed in Plant Surveillance Instructions.

1. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

2. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.6.H-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.6.H-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment No. 183.

LAI



AI

4.6.H. Snubbers

3. Visual Inspection Acceptance Criteria

Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.6.H.5. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the LIMITING CONDITIONS FOR OPERATION shall be met.

LAI

(A1)

4.6.H Snubbers

4.6.H.3 (Cont'd)

Additionally, snubbers attached to sections of safety-related systems that have experienced unexpected potentially damaging transients since the last inspection period shall be evaluated for the possibility of concealed damage and functionally tested, if applicable, to confirm OPERABILITY. Snubbers which have been made inoperable as the result of unexpected transients, isolated damage, or other random events, when the provisions of 4.6.H.7 and 4.6.H.8 have been met and any other appropriate corrective action implemented, shall not be counted in determining the next visual inspection interval.

(LAI)





4.6.H. Snubbers

4. FUNCTIONAL TEST Schedule, Lot Size, and Composition

During each refueling outage, a representative sample of 10% of the total of each type of safety-related snubbers in use in the plant shall be functionally tested either in place or in a bench test.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of snubbers within the types. The representative sample should be weighed to include more snubbers from severe service areas such as near heavy equipment.

The stroke setting and the security of fasteners for attachment of the snubbers to the component and to the snubber anchorage shall be verified on snubbers selected for FUNCTIONAL TESTS.

LAI

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~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(AI)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

4.6.H. Snubbers

5. FUNCTIONAL TEST Acceptance Criteria

The snubber FUNCTIONAL TEST shall verify that:

- a. Activation (restraining action) is achieved in both tension and compression within the specified range, except that inertia dependent, acceleration limiting mechanical snubbers may be tested to verify only that activation takes place in both directions of travel.
- b. Snubber bleed, or release where required, is present in both compression and tension within the specified range.
- c. For mechanical snubbers, the force required to initiate or maintain motion of the snubber is not great enough to overstress the attached piping or component during thermal movement, or to indicate impending failure of the snubber.
- d. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

(LAI)





(A1)

4.6.H. Snubbers

4.6.H.5 (Cont'd)

c. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

6. FUNCTIONAL TEST Failure Analysis and Additional Test Lots

(LAI) →

An engineering evaluation shall be made of each failure to meet the FUNCTIONAL TEST acceptance criteria to determine the cause of the failure. The result of this analysis shall be used, if applicable, in selecting snubbers to be tested in the subsequent lot in an effort to determine the OPERABILITY of other snubbers which may be subject to the same failure mode. Selection of snubbers for future testing may also be based on the failure analysis. For each snubber that does not meet the FUNCTIONAL TEST acceptance criteria, an additional lot equal to 10 percent of the remainder of that type of snubbers shall be functionally tested. Testing shall continue until no additional inoperable snubbers are found within subsequent lots or all snubbers of the original FUNCTIONAL TEST type have been tested or all suspect snubbers identified by the failure analysis have been tested, as applicable.



AI

4.6.H. Snubbers

4.6.H.6 (Cont'd)

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency, all snubbers of the same design subject to the same defect shall be functionally tested.

This testing requirement shall be independent of the requirements stated above for snubbers not meeting the FUNCTIONAL TEST acceptance criteria.

The discovery of loose or missing attachment fasteners will be evaluated to determine whether the cause may be localized or generic. The result of the evaluation will be used to select other suspect snubbers for verifying the attachment fasteners, as applicable.

7. FUNCTIONAL TEST Failure - Attached Component Analysis

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are restrained by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components restrained by the

LAI

(AI)

4.6.H. Snubbers

4.6.H.7 (Cont'd)

snubber(s) were adversely affected by the inoperability of the snubber(s), and in order to ensure that the restrained component remains capable of meeting the designed service.

8. Functional Testing Of Repaired and Spare Snubbers

(LAI)

Snubbers which fail the visual inspection or the FUNCTIONAL TEST acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the FUNCTIONAL TEST results shall meet the FUNCTIONAL TEST criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the FUNCTIONAL TEST must have been performed within 12 months before being installed in the unit.

9. Exemption from Visual Inspection or FUNCTIONAL TESTS

Permanent or other exemptions from visual inspections and/or functional testing for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and if applicable snubber life destructive testing was performed to qualify snubber OPERABILITY for the applicable design conditions at either the

(A1)

4.6.H. Snubbers

4.6.H.9 (Cont'd)

completion of their fabrication or at a subsequent date. Snubbers so exempted shall continue to be listed in the plant instructions with footnotes indicating the extent of the exemptions.

(LAI) →

10. Snubber Service Life Program

The service life of snubbers may be extended based on an evaluation of the records of FUNCTIONAL TESTS, maintenance history, and environmental conditions to which the snubbers have been exposed.



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BFN
Unit 3

3.6/4.6-23a

AMENDMENT NO. 134

LA1

Table 4.6.H-1

JUL 05 1994

SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS		
	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
50	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or more	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Column A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

JUL 05 1994

Table 4.6.H-1 (Continued)

SNUBBER VISUAL INSPECTION INTERVAL

- Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.
- Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C, but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.
- Note 6: The provisions of Specification 1.0.LL are applicable for all inspection intervals up to and including 48 months.

LAI

PAGE 18 OF 18



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

ADMINISTRATIVE

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 These surveillances are a duplication of the regulations found in 10 CFR 50 Appendix H. These regulations require licensee compliance and can not be revised by the licensee. Therefore, these details of the regulations within the Technical Specifications are repetitious and unnecessary. Furthermore, approved exemptions to the regulations, and exceptions presented within the regulations themselves, are also details which are adequately presented without repeating the details within the Technical Specifications. Therefore, retaining the requirement to meet the requirements of 10 CFR 50 Appendix H, as modified by approved exemptions, and eliminating the Technical Specification details that are also found in Appendix H, is considered a presentation preference which is administrative in nature.

- A3 For clarity, the terms "prior to and during startup" and "prior to" have been replaced with "15 minutes". This Frequency is effectively the same since the proposed Surveillance now must be performed no more than 15 minutes prior to startup of the idle recirculation loop. This is essentially equivalent to the current requirements.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

- A4 Proposed SR 3.4.9.4 requires verification that the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature are within 50°F of each other. CTS 3.6.A.6/4.6.A.6 requires verification that the temperatures between the idle and operating recirculation loops are within 50°F of each other. The temperature of the "operating recirculation loop" is considered equivalent to the RPV temperature. Therefore, this change is considered administrative.
- A6 Proposed SRs 3.4.9.5, 6 & 7 require the reactor vessel flange and head flange temperatures be verified > 82°F, while CTS 4.6.A.5 requires the reactor vessel shell temperature immediately below the head flange be recorded. The BFN procedure that implements this requirement requires the vessel flange and head flange temperature be verified and requires the shell temperature be recorded. Since the intent of the surveillance is to verify vessel flange and head flange temperature to satisfy CTS 3.6.A.5 and both the current and the proposed SRs do this, the two are considered equivalent. As such, the proposed change is administrative.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 A new Surveillance Requirement has been added. SR 3.4.9.2 ensures the RCS pressure and temperature are within the criticality limits once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality. This is an additional restriction on plant operation.
- M2 Three new Surveillance Requirements have been added. SR 3.4.9.5 ensures the vessel head is not tensioned at too low a temperature every 30 minutes. SRs 3.4.9.6 and 3.4.9.7 ensure the vessel and head flange temperatures do not exceed the minimum allowed temperature. These are additional restrictions on plant operation since the current requirements have no times specified.
- M3 ACTIONS have been added (proposed ACTIONS A, B, and C) to provide direction when the LCO is not met. Currently, no real ACTIONS are provided. These ACTIONS are consistent with the BWR Standard Technical Specification, NUREG 1433, and are additional restrictions on plant operation.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LAI Details of the methods for performing Surveillances, and any requirement to record data, are relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 The Frequency of this Surveillance has been changed from 15 minutes to 30 minutes. Verification that RCS temperature is within limits every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes is reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits a reasonable time for assessment and correction of minor deviations. In addition, this new Frequency is consistent with the BWR Standard Technical Specification, NUREG 1433.
- L2 The Frequency of this Surveillance has been changed from 15 minutes to 30 minutes. The metal temperature is not expected to change rapidly due to its large mass, thus a 30 minute Frequency is adequate. In addition, this new Frequency is consistent with the BWR Standard Technical Specification, NUREG 1433.

PAGE 3 OF 7



JUSTIFICATION FOR CHANGES
CTS 3.6.B/4.6.B - COOLANT CHEMISTRY

RELOCATED SPECIFICATIONS

R1 The chemistry limits are provided to prevent long term component degradation and provide long term maintenance of acceptable structural conditions of the system. The associated surveillances are not required to ensure immediate operability of the reactor coolant system. Therefore, the requirements specified in current Specification 3.6.B/4.6.B did not satisfy the NRC Final Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the Browns Ferry Unit 2 Technical Specifications and have been relocated to plant documents controlled in accordance with 10CFR50.59.

JUSTIFICATION FOR CHANGES
CTS 3.6.G/4.6.G - STRUCTURAL INTEGRITY

RELOCATED SPECIFICATIONS

RI The structural integrity inspections are provided to prevent long term component degradation and provide long term maintenance of acceptable structural conditions of the system. The associated inspections are not required to ensure immediate operability of the system. Therefore, the requirements specified in current Specification 3.6.G/4.6.G did not satisfy the NRC Final Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the BFN Unit 2 Technical Specifications and have been relocated to plant documents controlled in accordance with 10CFR50.59.



JUSTIFICATION FOR CHANGES
CTS 3.6.H/4.6.H - SNUBBERS

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LAI Snubber inspection requirements are part of the BFN Inservice Inspection (ISI) Program and are being relocated to the ISI program documents. Requirements for the ISI Program are specified in 10 CFR 50.55a to be performed in accordance with ASME Section XI. NRC regulations contain the necessary programmatic requirements for ISI without repeating them in the proposed BFN ISTS. Changes to the ISI Program are controlled in accordance with 10 CFR 50.59. With the removal of operability requirements from the Technical Specifications, snubber operability requirements will be determined in accordance with Technical Specification system operability requirements.

Section 3.4, Reactor Coolant System (RCS) Bases

The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of the proposed Browns Ferry Unit 2 Technical Specification Section 3.4, consistent with the BWR Standard Technical Specification, NUREG 1433. The revised Bases are as shown in the proposed Browns Ferry Unit 2 Technical Specification Bases.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



NOV 22 1988

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

(A1)

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and containment cooling systems.

Objective

To assure the OPERABILITY of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

~~A. Core Spray System (CSS)~~

(A1)

LCO 3.5.1

1. The CSS shall be OPERABLE:

(1) PRIOR TO STARTUP from a COLD CONDITION, or

(A6)

(2) when there is irradiated fuel in the vessel and when the reactor vessel pressure is greater than atmospheric pressure, except as specified in Specification 3.5.A.2.

Applicability

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the surveillance requirements of the core and containment cooling systems when the corresponding limiting condition for operation is in effect.

Objective

To verify the OPERABILITY of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

~~A. Core Spray System (CSS)~~

(A2)

1. Core Spray System Testing.

(L1) Item Actual or (A3) Frequency
SR 3.5.1.9 a. Simulated Automatic Actuation test
Once/18 mos Operating Cycle

Proposed Note to SR 3.5.1.9

(A4) SR 3.5.1.6 b. Pump OPERABILITY Per Specification 1.0.MM

(A9) c. Motor Operated Valve OPERABILITY Per Specification 1.0.MM

SR 3.5.1.6 d. System flow rate: Each loop shall deliver at least 6250 gpm against a system head corresponding to a (A3) Once/3-months (92 days)



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~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.A Core Spray System (CSS)~~

(A1)

~~4.5.A Core Spray System (CSS)~~

~~4.5.A.1.d (Cont'd)~~

2. ACTION A: If one CSS loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are OPERABLE. (AS)

ACTION H: (circled)

(M4) Be in Mode 3 in 12hrs

3. ACTION B & H: If Specification 3.5.A.1 or Specification 3.5.A.2 cannot be met, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours. (36) (L2)

4. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one OPERABLE pump and associated diesel generator shall be OPERABLE, except with the reactor vessel head removed as specified in 3.5.A.5 or PRIOR TO STARTUP as specified in 3.5.A.1. (A6)

See Justification for Changes for BFN 15TS 3.5.2

SR 3.5.1.6: 105 psi differential pressure between the reactor vessel and the primary containment.

(A9) e. Check Valve Per Specification 1.0.MM

SR 3.5.1.2

f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct position. (A7)

(A1) (A3) (31 days) (Once/Month)

2. No additional surveillance is required.

See Justification for Changes for BFN 15TS 3.8.1

(A7) *

Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

A2 1. The RHRS shall be OPERABLE #:

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

Applicability

A1

A6

1. a. Actual or Simulated Automatic Actuation Test

L1 A3 Once/18 months Operating Cycle

SR 3.5.1.6 b. Pump OPERABILITY

A3 Per Specification 1.0.MM

A9 c. Motor Operated valve OPERABILITY

A3 Per Specification 1.0.MM

SR 3.5.1.6 d. Pump Flow Rate

A3 Once/3 months 92 days

A9 e. Test Check Valve

A3 Per Specification 1.0.MM 31 days

SR 3.5.1.2 f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct position.

A3 Once/Month

SR 3.5.1.4 g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator.

A3 Once/Month

SR 3.5.1.2 Note

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

A7

AUG 02 1989

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENT~~

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B.1 (cont'd)~~
SR 3.5.1.6

M3

2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.

2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

See Justification for Changes for BFN 1STS 3.6.2.4

ACTION A

ACTION H

3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.

(A1)

~~3. No additional surveillance required.~~

(A5)

See Justification for Changes for BFN 1STS 3.8.1

ACTION B

4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(A1)

~~4. No additional surveillance required.~~

36

L2

M4

in Mode 3 in 12 hrs and



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHRG) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRG) (LPCI and Containment Cooling)~~

ACTIONS
B+H

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.
36 L2

(A1) 8. No additional surveillance required.

Be in Mode 3 in 12 hrs

(M4)

See Justification for Changes for BFN ISTS 3.5.2

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

(R1)

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

AUG 02 1989

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

SURVEILLANCE REQUIREMENTS

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(RI)

- 12. If one RHR pump or associated heat exchanger located on the unit cross-connection in the adjacent unit is inoperable for any reason (including valve inoperability, pipe break, etc.), the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pump and associated diesel generator are OPERABLE.
- 13. If RHR cross-connection flow or heat removal capability is lost, the unit may remain in operation for a period not to exceed 10 days unless such capability is restored.

- 12. No additional surveillance required.
- 13. No additional surveillance required.

(LAS)

14. All recirculation pump discharge valves shall be OPERABLE PRIOR TO STARTUP (or closed if permitted elsewhere in these specifications).

SR 3.5.1.5

- 14. All recirculation pump discharge valves shall be tested for OPERABILITY during any period of COLD SHUTDOWN CONDITION exceeding 48 hours, if OPERABILITY tests have not been performed during the preceding 31 days.

Note for SR 3.5.1.5

~~3.9/4.9 AUXILIARY ELECTRICAL SYSTEM~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.9.A. Auxiliary Electrical Equipment~~

~~4.9.A. Auxiliary Electrical System~~

See Justification for Changes for BFN ISTS Section 3.3.8.1

See Justification for Changes for BFN ISTS 3.8.7

5. Logic Systems

a. Common accident signal logic system is OPERABLE.

See Justification for Changes for BFN ISTS 3.8.1

b. 480-V load shedding logic system is OPERABLE.

6. There shall be a minimum of 35,280 gallons of diesel fuel in each of the 7-day diesel-generator fuel tank assemblies.

See Justification for Changes for BFN ISTS 3.8.3

4.9.A.4. (Cont'd)

c. The loss of voltage and degraded voltage relays which start the diesel generators from the 4-kV shutdown boards shall be calibrated annually for trip and reset and the measurements logged. These relays shall be calibrated as specified in Table 4.9.A.4.c.

d. 4-kV shutdown board voltages shall be recorded once every 12 hours.

5. 480-V RMOV Boards 1D and 1E

SR 3.5.1.12

a. Once per ^{12 mo.} operating cycle the automatic transfer feature for 480-V RMOV boards 1D and 1E shall be functionally tested to verify auto-transfer capability. (A3)

~~3.9/4.9 AUXILIARY ELECTRICAL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.9.A. ~~Auxiliary Electrical Equipment~~

4.9.A. ~~Auxiliary Electrical System~~

~~3.9.A.3 (Cont'd)~~

- d. The 480-V shutdown boards 1A and 1B are energized.
- e. The units 1 and 2 diesel auxiliary boards are energized.

See Justification for Changes for BFN ISTS 3.8.7

- f. Loss of voltage and degraded voltage relays OPERABLE on 4-kV shutdown boards A, B, C, and D.

See Justification for Changes for BFN ISTS 3.3.8.1

- g. Shutdown buses 1 and 2 energized.

See Justification for Changes for BFN ISTS 3.8.1

(A13)

- h. The 480-V reactor motor-operated valve (RMOV) boards 1D & 1E are energized with motor-generator (mg) sets 1DN, 1DA, 1EN, and 1EA in service.

- 4. The three 250-V unit batteries, the four shutdown board batteries, a battery charger for each battery, and associated battery boards are OPERABLE.

See Justification for Changes for BFN ISTS 3.8.4 and 3.8.7

4. Undervoltage Relays

- a. (Deleted)
- b. Once every 18 months, the conditions under which the loss of voltage and degraded voltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator will start.

~~3.9/4.9 AUXILIARY ELECTRICAL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

A1

~~SURVEILLANCE REQUIREMENTS~~

~~3.9.B. Operation With Inoperable Equipment~~

12. When one 480-V shutdown board is found to be INOPERABLE, the reactor will be placed in HOT STANDBY CONDITION within 12 hours and COLD SHUTDOWN CONDITION within 24 hours.

See Justification for Changes for BFN 15TS 3.8.7

13. If one 480-V RMOV board mg set is INOPERABLE, REACTOR POWER OPERATION may continue for a period not to exceed seven days, provided the remaining 480-V RMOV board mg sets and their associated loads remain OPERABLE.

A13

14. If any two 480-V RMOV board mg sets become INOPERABLE, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

15. If the requirements for operating in the conditions specified by 3.9.B.1 through 3.9.B.14 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

See Justification for Changes for BFN 15TS Section 3.8

FEB 07 1991

~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

3.5.D Equipment Area Coolers

1. The equipment area cooler associated with each RHK pump and the equipment area cooler associated with each set of core spray pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.
2. When an equipment area cooler is not OPERABLE, the pump(s) served by that cooler must be considered inoperable for Technical Specification purposes.

4.5.D Equipment Area Coolers

1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

LAS

~~F. High Pressure Coolant Injection System (HPCIS)~~ (A2)

- LCO 3.5.1
- Applicability (A11)
- Proposed Note for SR 3.5.1.8
1. The HPCI system shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in Specification 3.5.E.2. OPERABILITY shall be determined within 12 hours after reactor steam pressure reaches 150 psig from a GOLD CONDITION, or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

L A 6

(A1) ~~F. High Pressure Coolant Injection System (HPCIS)~~

1. HPCI Subsystem testing shall be performed as follows:
 - a. Simulated Automatic Actuation Test (Actual or (L1)) Once/18 months (A3)
 - b. Pump OPERABILITY (SR 3.5.1.7) Per Specification 1.0.MM (A4)
 - c. Motor Operated Valve OPERABILITY (A9) Per Specification 1.0.MM (L11)
 - d. Flow Rate at normal reactor vessel operating pressure (SR 3.5.1.7) Once/3-months (A3) 92 days (920 to 1010 psig) (L9)
- Proposed Note for SR 3.5.1.9
- Proposed Note for SR 3.5.1.7

(A10)

FEB 07 1991

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.E High Pressure Coolant Injection System (HPCIS)~~

~~4.5.E High Pressure Coolant Injection System (HPCIS)~~

~~4.5.E.1 (Cont'd)~~

SR 3.5.1.8 e. Flow Rate at Once/18
150 psig months

≤ 165 psig
(L10)

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

SR 3.5.1.2

f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct position.

31 days
Once/31 days
(A3)

(A1) ~~2. No additional surveillances are required.~~

2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ~~ABS, CSS, RHRS~~ (LPCI), and RCICS are OPERABLE.

(L4)

verified immediately

3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

(A12)

62 in 3 in 12 hrs

(24) (36) (L2)

ACTIONS G+H

Action H (A5)
Proposed Action D (L3)

(A7)
* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

(M4)

F. Reactor Core Isolation Cooling System (RCICS)

F. Reactor Core Isolation Cooling System (RCICS)

1. The RCICS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

1. RCIC Subsystem testing shall be performed as follows:
a. Simulated Auto- Once/18
matic Actuation months
Test

BFN Unit 1

See Justification for Changes 3.5/4.5-14 for BFN ISTS 3.5.3

AMENDMENT NO. 180



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.G Automatic Depressurization System (ADS)~~

~~4.5.G Automatic Depressurization System (ADS)~~ (A3)

(A2) 1. Six valves of the Automatic Depressurization System shall be OPERABLE:

1. During each operating cycle the following tests shall be performed on the ADS:

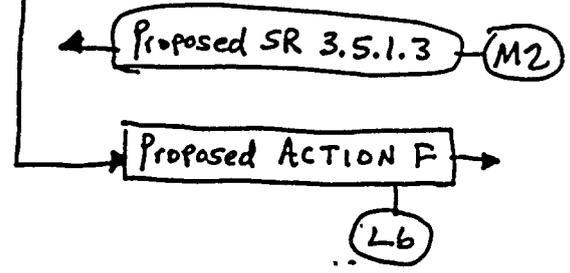
Applicability (A11) (1) PRIOR TO STARTUP from a COLD CONDITION, or, (L5) (150) (2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.G.2 and 3.5.G.3 below.

SR 3.5.1.10 a. (L1) Actual or (L1) A simulated automatic actuation test shall be performed PRIOR TO STARTUP after each refueling outage. (A3) Manual surveillance of the relief valves is covered in 4.6.D.2. (A8) (A4) Proposed Note to SR 3.5.1.10

ACTIONS E 2. With one of the above required ADS valves inoperable, provided the HPCI system, the core spray system, and the LPCI system are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least a HOT SHUTDOWN CONDITION within the next 12 hours and reduce reactor steam dome pressure to ≤ 105 psig within 24 hours. (36) (L2) (150) (L5)

(A1) 2. No additional surveillances are required.

ACTION G+H 3. With two or more of the above required ADS valves inoperable, be in at least a HOT SHUTDOWN CONDITION within 12 hours and reduce reactor steam dome pressure to ≤ 105 psig within 24 hours. (36) (L2) (150) (L5)



ACTIONS G Required Action H.1 (LEO 3.0.3) (M1) MODE 2 within 7hrs (L12) (for ADS only) MODE 3 within 13hrs (L12) MODE 4 within 37hrs (L12)

DEC 07 1994

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.C Coolant Leakage

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

3.6.D Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

See Justification for Changes for BFN 1STS 3.4.3

4.6.C Coolant Leakage

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

See Justification for Changes for BFN 1STS 3.4.5

4.6.D Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

SE 3.5.1.11

Proposed Note for FR 3.5.1.11

LB

(A1)

MAY 19 1994

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.H. Maintenance of Filled Discharge Pipe~~

~~4.5.H. Maintenance of Filled Discharge Pipe~~

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSS head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

LA2

See Justification for Changes for BFN 15 TS 3.5.3

SR 3.5.1.1

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

LA7

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined.

LA3

2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

LA4

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

LA3

4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

(A1)

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and containment cooling systems.

Objective

To assure the OPERABILITY of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

(A1)

LCO 3.5.1

1. The CSS shall be OPERABLE:

(A2)

(1) PRIOR TO STARTUP from a COLD CONDITION, or

(A6)

Applicability

(2) when there is irradiated fuel in the vessel and when the reactor vessel pressure is greater than atmospheric pressure, except as specified in Specification 3.5.A.2.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the surveillance requirements of the core and containment cooling systems when the corresponding limiting condition for operation is in effect.

Objective

To verify the OPERABILITY of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. Core Spray System Testing.

(L1)

Item

Frequency

SR 3.5.1.9

Actual or

(Simulated Automatic Actuation test

Once/18 mos. Operating Cycle

(A3)

Proposed Note to SR 3.5.1.9

(A4)

SR 3.5.1.6

b. Pump Operability

Per Specification 1.0.MM

(A3)

(A9)

c. Motor Operated Valve OPERABILITY

Per Specification 1.0.MM

SR 3.5.1.6

d. System flow rate: Each loop shall deliver at least 6250 gpm against a system head corresponding to a

Once/3-months

(A3)

92 days

2... 15



AUG 02 1989

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.A Core Spray System (CSS) (A1)

4.5.A Core Spray System (CSS)

4.5.A.1.d (Cont'd)

2. ACTION A
ACTION H

If one CSS loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are OPERABLE.

SR 3.5.1.6

105 psi differential pressure between the reactor vessel and the primary containment.

(A9) e. Check Valve Per Specification 1.0.MM

SR 3.5.1.2

1. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct position.

Once/Month (A3) 31 days (A7)

3. ACTION B + H

If Specification 3.5.A.1 or Specification 3.5.A.2 cannot be met, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(A1) 2. No additional surveillance is required.

See Justification for Changes for BFN ISTS 3.8.1

4. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one OPERABLE pump and associated diesel generator shall be OPERABLE, except with the reactor vessel head removed as specified in 3.5.A.5 or PRIOR TO STARTUP as specified in 3.5.A.1.

(A7)

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

See Justification for Changes for BFN ISTS 3.5.2

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS APR 19 1994

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

A2 1. The RHRS shall be OPERABLE #:

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

Applicability

1. a. Actual or L1 (A3) Simulated Automatic Actuation Test Once/18 mos Operating Cycle

SR 3.5.1.6 b. Pump OPERABILITY Per Specification 1.0.MM (A3)

(A9) c. Motor Operated valve OPERABILITY Per Specification 1.0.MM

SR 3.5.1.6 d. Pump Flow Rate Once/3 months (A3) 92 days

(A9) e. Testable Check Valve Per Specification 1.0.MM

SR 3.5.1.2 f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct position. (A7) Once/Month (A3) 31 days

SR 3.5.1.4 g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. (A3) Once/Month

SR 3.5.1.2 Note

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

(A7) Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



AUG 02 1989

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENT

3.5.B ~~Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

4.5.B. ~~Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

4.5.B.1 (cont'd)

SR 3.5.1.6

M3

2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.

2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

See Justification for Changes for BFN 1STS 3.6.2.4

ACTION A

3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days

L7

(A1)

3. No additional surveillance required.

ACTION H

provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.

(AS)

See Justification for Changes for BFN 1STS 2.8.1

ACTION B

4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

L7

(A1)

4. No additional surveillance required.

24 36

MIDE 3 in 12 hrs

L2



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

ACTIONS
B+H

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, ~~an orderly shutdown shall be initiated~~ and the reactor shall be placed in the COLD SHUTDOWN CONDITION within ~~24~~³⁶ hours. (L2)

~~4.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

~~8. No additional surveillance required.~~

Be in Mode 3 in 12 hrs (M4)

See Justification for Changes for BFN 1573 3.5.2

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

(R1) 11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

~~4.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

12. If three RHR pumps or associated heat exchangers located on the unit cross-connection in the adjacent units are inoperable for any reason (including valve inoperability, pipe break, etc.), the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pump and associated diesel generator are OPERABLE.

(R1)

13. If RHR cross-connection flow or heat removal capability is lost, the unit may remain in operation for a period not to exceed 10 days unless such capability is restored.

14. All recirculation pump discharge valves shall be OPERABLE PRIOR TO STARTUP (or closed if permitted elsewhere in these specifications).

(LAS)

~~12. No additional surveillance required.~~

~~13. No additional surveillance required.~~

SR 3.5.1.5

14. All recirculation pump discharge valves shall be tested for OPERABILITY - (during any period of COLD SHUTDOWN CONDITION exceeding 48 hours, if OPERABILITY tests have not been performed during the preceding 31 days.

Note for SR 3.5.1.5.

NOV 04 1991

~~3.9/4.9 AUXILIARY ELECTRICAL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.9.A Auxiliary Electrical Equipment~~

~~4.9.A Auxiliary Electrical System~~

~~3.9.A.3. (Cont'd)~~

- d. The 480-V shutdown boards 2A and 2B are energized.
- e. The units 1 and 2 diesel auxiliary boards are energized.

See Justification for Changes for BFN ISTS 3.8.7

- f. Loss of voltage and degraded voltage relays OPERABLE on 4-kV shutdown boards A, B, C, and D.

See Justification for Changes for BFN ISTS 3.3.8.1

- g. Shutdown buses 1 and 2 energized.

See Justification for Changes for BFN ISTS 3.8.1

- h. The 480-V reactor motor-operated valve (RMOV) boards 2D & 2E are energized with motor-generator (mg) sets 2DN, 2DA, 2EN, and 2EA in service.

(A13)

- 4. The three 250-V unit batteries, the four shutdown board batteries, a battery charger for each battery, and associated battery boards are OPERABLE.

See Justification for Changes for BFN ISTS 3.8.4

- 4. Undervoltage Relays
 - a. (Deleted)
 - b. Once every 18 months, the conditions under which the loss of voltage and degraded voltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator will start.

FEB 12 1991

~~3.9/4.9 AUXILIARY ELECTRICAL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.9.A. Auxiliary Electrical Equipment~~

~~4.9.A. Auxiliary Electrical System~~

See Justification for Changes for BFN ISTS Section 3.3.8.1

See Justification for Changes for BFN ISTS 3.8.7

5. Logic Systems
a. Common accident signal logic system is OPERABLE.

See Justification for Changes for BFN ISTS 3.8.1

b. 480-V load shedding logic system is OPERABLE.

6. There shall be a minimum of 35,280 gallons of diesel fuel in each of the 7-day diesel-generator fuel tank assemblies.

See Justification for Changes for BFN ISTS 3.8.3

4.9.A.4. (Cont'd)

c. The loss of voltage and degraded voltage relays which start the diesel generators from the 4-kV shutdown boards shall be calibrated annually for trip and reset and the measurements logged. These relays shall be calibrated as specified in Table 4.9.A.4.c.

d. 4-kV shutdown board voltages shall be recorded once every 12 hours.

5. 480-V RMOV Boards 2D and 2E
SR 3.5.1.12

a. Once per ^{18 mos.} operating cycle the automatic transfer feature for 480-V RMOV boards 2D and 2E shall be functionally tested to verify auto-transfer capability. (A3)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.9.B. Operation With Inoperable Equipment

A1

12. When one 480-V shutdown board is found to be INOPERABLE, the reactor will be placed in the HOT STANDBY CONDITION within 12 hours and COLD SHUTDOWN CONDITION within 24 hours.

See Justification for Changes for BFN ISTS 3.8.7

13. If one 480-V RMOV board mg set is INOPERABLE, REACTOR POWER OPERATION may continue for a period not to exceed seven days, provided the remaining 480-V RMOV board mg sets and their associated loads remain OPERABLE.

A13

14. If any two 480-V RMOV board mg sets become INOPERABLE, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

15. If the requirements for operating in the conditions specified by 3.9.B.1 through 3.9.B.14 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

See Justification for Changes for BFN ISTS Section 3.8



FEB 07 1991

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.5.D Equipment Area Coolers

1. The equipment area cooler associated with each RHR pump and the equipment area cooler associated with each set of core spray pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.
2. When an equipment area cooler is not OPERABLE, the pump(s) served by that cooler must be considered inoperable for Technical Specification purposes.

4.5.D Equipment Area Coolers

1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

(LAS)

~~E. High Pressure Coolant Injection System (HPCIS)~~

(A2)

LCO 3.5.1

Applicability (All)

Proposed Note for SR 3.5.1.8

1. The HPCI system shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in Specification 3.5.E.2. OPERABILITY shall be determined within 12 hours after reactor steam pressure + flow reaches 150 psig from a COLD CONDITION, or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

(LAG)

(A1)

~~F. High Pressure Coolant Injection System (HPCIS)~~

SR 3.5.1.9

Proposed Note for SR 3.5.1.9

(A4)

SR 3.5.1.7

(L11)

Proposed Note for SR 3.5.1.7

(A10)

1. HPCI Subsystem testing shall be performed as follows:

Actual or (L1)

- a. Simulated Automatic Actuation Test

Once/18 months

- b. Pump OPERABILITY

Per Specification 1.0.MM (A3)

- c. Motor Operated Valve OPERABILITY

Per Specification 1.0.MM (A9)

- d. Flow Rate at normal reactor vessel operating pressure

Once/3-months (A3)

(92 days)

920 to 1010 psig (L9)



LIMITING CONDITIONS FOR OPERATION (A1) SURVEILLANCE REQUIREMENTS

3.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

SR 3.5.L8 e. Flow Rate at Once/18
150 psig months

≤ 165 psig
L10
The HPCI pump shall deliver at least 5000 gpm during each flow rate test. 31 days

SR 3.5.1.2 f. Verify that Once/Month
each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct position. A3

(A1) 2. ~~No additional surveillances are required.~~

L4
2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 147 days, provided the CSS, RHRS (LPCI) and RCICS are OPERABLE and verified immediately.

Proposed ACTION D
L3
3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24-36 hours. Be in Mode 3 in 12 hrs

A5 ACTION H

A12
A7
* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

F. Reactor Core Isolation Cooling System (RCICS)

F. Reactor Core Isolation Cooling System (RCICS)

1. The RCICS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

1. RCIC Subsystem testing shall be performed as follows:
a. Simulated Auto- Once/18 matic Actuation months Test

BFN Unit 2

3.5/4.5-14

AMENDMENT NO. 190

See Justification for Changes for BFN 15TS 3.5.3



LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.G Automatic Depressurization System (ADS) (A1)

4.5.G Automatic Depressurization System (ADS) (A3)

1. Six valves of the Automatic Depressurization System shall be OPERABLE: (A2)

1. During each operating cycle the following tests shall be performed on the ADS: (L1)

(1) PRIOR TO STARTUP from a COLD CONDITION, or, (A11)

SR 3.5.1.10

a. A simulated automatic actuation test shall be performed PRIOR TO STARTUP after each refueling outage. (A3)

Applicability

(2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except in the COLD SHUT-DOWN CONDITION or as specified in 3.5.G.2 and 3.5.G.3 below. (150) (LS)

every 18 mos.

Manual surveillance of the relief valves is covered in 4.6.D.2. (A8)

Proposed Note to SR 3.5.1.10 (A4)

2. With one of the above required ADS valves inoperable, provided the HPCI system, the core spray system, and the LPCI system are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least a HOT SHUTDOWN CONDITION within the next 12 hours and reduce reactor steam dome pressure to 105 psig within hours. (150) (LS)

2. No additional surveillances are required. (A1)

ACTIONS E

Proposed SR 3.5.1.3 (M2)

ACTION G+H

3. With two or more of the above required ADS valves inoperable, be in at least a HOT SHUTDOWN CONDITION within 12 hours and reduce reactor steam dome pressure to 105 psig within hours. (150) (LS) (24) (36) (L2)

Proposed ACTION F (L6)

ACTIONS G

3. With two or more of the above required ADS valves inoperable, be in at least a HOT SHUTDOWN CONDITION within 12 hours and reduce reactor steam dome pressure to 105 psig within hours. (150) (LS) (24) (36) (L2)

Required Action H.1 (CCO 3.03) (M1)

Mode 2 within 7 hrs

Mode 3 within 13 hrs

Mode 4 within 37 hrs (L12)

(L12) (for ADS only)

AMENDMENT NO. 190



3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.C Coolant Leakage

- 2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

SR 89-77B/77A-2-039

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

- 3. If the condition in 1. or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

3.6.D Relief Valves

- 1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

See Justification for Changes for BFN 15TS 2.4.3

4.6.C Coolant Leakage

- 2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

See Justification for Changes for BFN 15TS 3.4.5

4.6.D Relief Valves

- 1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.
- 2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until

thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

SR 3.5.1.11

Proposed Note for SR 3.5.1.11

(L8)

(LA3)



AUG 02 1989

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.5.H. Maintenance of Filled Discharge Pipe

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

LA2

See Justification for Changes for BFN 15TS 3.5.3

4.5.H. Maintenance of Filled Discharge Pipe

SR 3.5.1.1

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

LA7

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined.

LA3

2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

LA4

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

LA3

4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

UNIT 3
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and containment cooling systems.

Applicability

Applies to the surveillance requirements of the core and containment cooling systems when the corresponding limiting condition for operation is in effect.

Objective

To assure the OPERABILITY of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Objective

To verify the OPERABILITY of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

Specification

~~A. Core Spray System (CSS)~~

~~A. Core Spray System (CSS)~~

LCO 3.5.1

1. The CSS shall be OPERABLE:

1. Core Spray System Testing.

- (1) PRIOR TO STARTUP from a COLD CONDITION, or
- (2) when there is irradiated fuel in the vessel and when the reactor vessel pressure is greater than atmospheric pressure, except as specified in Specification 3.5.A.2.

Applicability

Item	Frequency
(L1) Actual or Simulated Automatic Actuation test	Once/18 months Operating Cycle
SR 3.5.1.9	
Proposed Note to SR 3.5.1.9	
(A4) SR 3.5.1.6 } b. Pump OPERABILITY	Per Specification 1.0.MM
(A9) c. Motor Operated Valve OPERABILITY	Per Specification 1.0.MM
SR 3.5.1.6 } d. System flow rate: Each loop shall deliver at least 6250 gpm against a system head corresponding to a	Once/3 months 92 days



LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.A Core Spray System (CSS)

(A1)

- 2. **Action A** If one CSS loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are OPERABLE.
- Action H**

M4

Be in Mode 3 IN 12hrs

Action B+H

- 3. If Specification 3.5.A.1 or Specification 3.5.A.2 cannot be met, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours. (36) (L2)

4. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one OPERABLE pump and associated diesel generator shall be OPERABLE, except with the reactor vessel head removed as specified in 3.5.A.5 or PRIOR TO STARTUP as specified in 3.5.A.1.

(A6)

See Justification for changes for BFN ISTS 3.5.2

4.5.A Core Spray System (CSS)

4.5.A.1.d (Cont'd)

SR 3.5.1.6 } 105 psi differential pressure between the reactor vessel and the primary containment.

(A9) e. Testable Per Check Valve Specification 1.0MM

SR 3.5.1.2

f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct position.

(A3) Once/Month
31 days

(A5)

(A1)

2. No additional surveillance is required.

See Justification for Changes for BFN ISTS 3.8.1

(A7)

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A2) 1. The RHRS shall be OPERABLE #:

(1) PRIOR TO STARTUP from a COLD CONDITION; or (A6)

Applicability (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

1. a. Actual OR (L1) Simulated Automatic Actuation Test (A3) Once/18^{mos} Operating Cycle
- SR 3.5.1.6 b. Pump OPERABILITY (A3) Per Specification 1.0.MM
- (A9) c. Motor Operated valve OPERABILITY (A3) Per Specification 1.0.MM
- SR 3.5.1.6 d. Pump Flow Rate (A3) Once/3 months (92 days)
- (A9) e. Testable Check Valve (A3) Per Specification 1.0.MM
- SR 3.5.1.2 f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct position. (A7) Once/31^{days} Month
- SR 3.5.1.4 8. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. (A3) Once/31^{days} Month

(L13) OR verify the manual shutoff valve in the LPCI cross tie is closed

SR 3.5.1.2 Note

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

(A7) Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



AUG 02 1989

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

SURVEILLANCE REQUIREMENT

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B.1 (cont'd)~~

SR 3.5.1.6

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 gpm against an indicated system pressure of 250 psig.

M3

2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps—containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.

2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

See Justification for Changes for BFN ISTS 3.6.2.4

Action A

3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days

Action H

provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.

(A5)

See Justification for Changes for BFN ISTS 3.8.1

Action B

4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

24
36

L2

Be in mode 3 in 12 hrs

M4

(A1)

~~4. No additional surveillance required.~~



(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

Actions B+H

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, ~~an orderly shutdown shall be initiated~~ and the reactor shall be placed in the COLD SHUTDOWN CONDITION within ~~24~~ ³⁰ hours. (L2)

(A1)

~~8. No additional surveillance required.~~

Be in Mode 3 in 12 hrs

(M4)

See Justification For Changes for BFN ISTS 3.5.2

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

(R)

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The B and D RHR pumps on unit 2 which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0:MM when the cross-connect capability is required.



AUG 02 1989

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.B Residual Heat Removal System
(RHR) (LPCI and Containment
Cooling)~~

~~A.5.B Residual Heat Removal System
(RHR) (LPCI and Containment
Cooling)~~

12. If one RHR pump or associated heat exchanger located on the unit cross-connection in unit 2 is inoperable for any reason (including valve inoperability, pipe break, etc.), the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pump and associated diesel generator are OPERABLE.

13. If RHR cross-connection flow or heat removal capability is lost, the unit may remain in operation for a period not to exceed 10 days unless such capability is restored.

14. All recirculation pump discharge valves shall be OPERABLE PRIOR TO STARTUP (or closed if permitted elsewhere in these specifications).

~~12. No additional surveillance required.~~

13. No additional surveillance required.

SR 3.5.15

14. All recirculation pump discharge valves shall be tested for OPERABIL during any period of COLD SHUTDOWN CONDITION exceeding 48 hours, if OPERABILITY tests have not been performed during the preceding 31 days.

Note for SR 3.5.15

(RI)

(A1)

(A1)

(LAS)



FEB 14 1995

~~3.9/4.9 AUXILIARY ELECTRICAL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

~~3.9.A. Auxiliary Electrical Equipment~~

~~4.9.A. Auxiliary Electrical System~~

See Justification for changes for BFN ISTS Section 3.3.8.1

See Justification for changes for BFN ISTS 3.8.7

4.9.A.4. (Cont'd)

c. The loss of voltage and degraded voltage relays which start the diesel generators from the 4-kV shutdown boards shall be calibrated annually for trip and reset and the measurements logged. These relays shall be calibrated as specified in Table 4.9.A.4.c.

d. 4-kV shutdown board voltages shall be recorded once every 12 hours.

5. Logic Systems

- a. Accident signal logic system is OPERABLE.
- b. 480-volt load shedding logic system is OPERABLE.

See Justification for changes for BFN ISTS 3.8.1

6. There shall be a minimum of 35,280 gallons of diesel fuel in each of the 7-day diesel-generator fuel tank assemblies.

See Justification for changes for BFN ISTS 3.8.3

5. 480-V RMOV Boards 3D and 3E

(18 mos.) (A3)

SR 3.5.1.12

a. Once per operating cycle, the automatic transfer feature for 480-V RMOV boards 3D and 3E shall be functionally tested to verify auto-transfer capability.



(A1)

NOV 04 1991

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.9.A Auxiliary Electrical Equipment~~

~~4.9.A. Auxiliary Electrical System~~

~~3.9.A.3. (Cont'd)~~

e. Loss of voltage and degraded voltage relays OPERABLE on 4-kV shutdown boards 3EA, 3EB, 3EC, and 3ED.

See Justification for Changes for BFN ISTS 3.3.8.1

f. The 480-V diesel auxiliary boards 3EA and 3EB are energized.

See Justification for Changes for BFN ISTS 3.8.7

g. The 480-V reactor motor-operated valve (RMOV) boards 3D & 3E are energized with motor-generator (mg) sets 3DN, 3DA, 3EN, and 3EA in service.

(A13)

4. The 250-V shutdown board 3EB battery, all three unit batteries, a battery charger for each battery, and associated battery boards are OPERABLE.

4. Undervoltage Relays

a. (Deleted)

b. Once every 18 months, the conditions under which the loss of voltage and degraded voltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator will start.

See Justification for Changes for BFN ISTS 3.8.4 + 3.8.7

See Justification for Changes for BFN ISTS 3.8.1



~~3.9/4.9 AUXILIARY ELECTRICAL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.9.B Operation With Inoperable Equipment~~

10. When one 480-V shutdown board is found to be inoperable, the reactor will be placed in HOT STANDBY CONDITION within 12 hours and COLD SHUTDOWN CONDITION within 24 hours.

Justification for changes for BFN 1STS 3.8.7

11. If one 480-V RMOV board mg set is inoperable, REACTOR POWER OPERATION may continue for a period not to exceed seven days, provided the remaining 480-V RMOV board mg sets and their associated loads remain OPERABLE.

(A13)

12. If any two 480-V RMOV board mg sets become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

13. If the requirements for operation in the conditions specified by 3.9.B.1 through 3.9.B.12 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

See Justification for Changes for BFN 1STS section 3.8



(A1)

3.5.D Equipment Area Coolers

1. The equipment area cooler associated with each RHR pump and the equipment area cooler associated with each set of core spray pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.
2. When an equipment area cooler is not OPERABLE, the pump(s) served by that cooler must be considered inoperable for Technical Specification purposes.

4.5.D Equipment Area Coolers

1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

(LA5)

~~F. High Pressure Coolant Injection System (HPCIS)~~

(A2)

LCO 3.5.1

Applicability

(A11)

1. The HPCI system shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in Specification 3.5.E.2. OPERABILITY shall be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION, or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

Proposed Note for SR 3.5.1.8

(LAG)

(A1)

~~F. High Pressure Coolant Injection System (HPCIS)~~

SR 3.5.1.9

Proposed Note for SR 3.5.1.9

(A4)

(L11)

1. HPCI Subsystem testing shall be performed as follows:

a. ^(Actual or) Simulated Automatic Actuation Test

Once/18 months

b. Pump OPERABILITY

Per Specification 1.0.MM

c. Motor Operated Valve OPERABILITY

Per Specification 1.0.MM

d. Flow Rate at normal reactor vessel operating pressure

Once/3-months

92 days

Proposed Note for SR 3.5.1.7

(A10)

920 to 1010 psig

(L9)



(A1)

3.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

SR 3.5.1.8

e. Flow Rate at Once/18 months

≤ 165 psig

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

L10

f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct position.

SR 3.5.1.2

2. No additional surveillances are required.

2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS (LPCI), and RCICS are OPERABLE immediately.

3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24-36 hours.

Proposed ACTION D

Actions G+H

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

F. Reactor Core Isolation Cooling System (RCICS)

F. Reactor Core Isolation Cooling System (RCICS)

1. The RCICS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

1. RCIC Subsystem testing shall be performed as follows:

a. Simulated Auto- Once/18 matic Actuation months Test

BFN Unit 3

See Justification for Changes 3.5/4.5-14 for BFN 1STS 3.5.3

AMENDMENT NO. 152

MAY 19 1994

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.G Automatic Depressurization System (ADS)

4.5.G Automatic Depressurization System (ADS)

1. Six valves of the Automatic Depressurization System shall be OPERABLE:

(1) PRIOR TO STARTUP from a COLD CONDITION, or,
(2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.G.2 and 3.5.G.3 below.

Applicability
All

2. With one of the above required ADS valves inoperable, provided the HPCI system, the core spray system, and the LPCI system are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least a HOT SHUTDOWN CONDITION within the next 12 hours and reduce reactor steam dome pressure to ≤ 105 psig within 24 hours.

Actions
F

Action
G & H

3. With two or more of the above required ADS valves inoperable, be in at least a HOT SHUTDOWN CONDITION within 12 hours and reduce reactor steam dome pressure to ≤ 105 psig within 24 hours.

Action
G

(for ADS only)
L12

Required Action H1 (LCO 3.0.3)
mode 2 within 7 hrs
mode 3 within 13 hrs
mode 4 within 37 hours

1. During each operating cycle the following tests shall be performed on the ADS:

SR 3.5.1.10 a. A simulated automatic actuation test shall be performed PRIOR TO STARTUP after each refueling outage. Manual surveillance of the relief valves is covered in 4.6.D.2.

Proposed Note to SR 3.5.1.10

2. No additional surveillances are required.

Proposed SR 3.5.1.3

Proposed Action F

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.C Coolant Leakage

- 2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

- 3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

3.6.D. Relief Valves

- 1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

See Justification for Changes for BFN ISTS 3.4.3

4.6.C Coolant Leakage

- 2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

See Justification for Changes for BFN ISTS 3.4.5

4.6.D. Relief Valves

- 1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.
- 2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened

until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve

SR3.5.1.11

Proposed Note for SR 3.5.1.11

LB

AMENDMENT NO. 188

PAGE 14 OF 15



(A1)

~~3.5.H Maintenance of Filled Discharge Pipe~~

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

LA2

See Justification for Changes for BFN 15 TS 3.5.3

~~4.5.H. Maintenance of Filled Discharge Pipe~~

SR 3.5.1.1

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

LA7

1. Every month and prior to the testing of the RHRs (LPCI and Containment Spray) and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow determined.

LA3

2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

LA4

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

LA3

4. When the RHRs and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 Five current LCOs, 3.5.A, 3.5.B, 3.5.E, 3.5.G, and 3.5.H, have been combined into one proposed LCO (3.5.1). As such, the new LCO combines the three ECCS spray/injection Systems (HPCI, LPCI, and CS) into one LCO statement. The Bases continue to describe what components make up an ECCS subsystem. The new LCO statement also specifies that the six ADS valves are required. In addition, the ADS valve cycling requirements located in current Specification 4.6.D.1 are included as part of ADS operability. Thus, if an ADS valve does not cycle, the affected ECCS system is considered inoperable and the appropriate ACTION taken.

- A3 The Frequencies of "Once/operating cycle," "during each operating cycle," and "after each refueling outage" have been changed to "18 months." This is considered equivalent since 18 months is the length of an operating cycle or a refueling outage cycle. The Frequencies of "Once/3 months" and "Per Specification 1.0.MM" have been changed to "92 days," or "In accordance with the Inservice Testing program" as appropriate. The IST program test frequency for pumps is every 3 months and is currently defined by Specification 1.0.MM. Therefore, this change is considered administrative in nature. The Frequency of "Once/month" has been changed to "31 days."

- A4 Notes allowing actual vessel injection or ADS valve actuation to be excluded from this test (simulated automatic actuation test) have been added to proposed SR 3.5.1.9 and SR 3.5.1.10. Since the current



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING

requirements state the test is "simulated" (i.e., valve actuation and vessel injection are inherently excluded), this allowance is considered administrative in nature.

- A5 Proposed Condition H provides direction for various interrelationships between HPCI and ADS, and between LPCI and CS. The Action requires entry into LCO 3.0.3 for various combinations of inoperability which are consistent with the present required actions for the same various combinations. The actual requirements are not being changed.
- A6 The existing Applicability for Core Spray System (CSS) Operability (3.5.A.1), and Low Pressure Coolant Injection (LPCI) Operability (3.5.B.1), requires both systems to be Operable whenever irradiated fuel is in the vessel and prior to startup from a COLD CONDITION. The proposed change (LCO 3.5.1 Applicability) requires them to be Operable in Modes 1, 2 and 3. This change more clearly defines the conditions when CSS and LPCI are required to be Operable without changing the specific requirements which are currently located in individual specifications for each system. This change is administrative because the same requirements for Operability currently listed in specific specifications will be labelled APPLICABILITY and apply to the entire ISTS Section 3.5.1, ECCS-Operating. The 3.5.A.2, 3.5.B.2, and 3.5.B.7 Applicabilities are only cross references and have been deleted.
- A7 The clarifying information contained in the "*" footnote has been moved to the proposed Bases for SR 3.5.1.2. The intent of the surveillance is to assure that the proper flow paths will exist for ECCS operation. The Bases clarifies that a valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. As such, moving this clarifying statement to the Bases is an administrative change.
- A8 This requirement has been deleted since it only provides reference to another Specification, and does not provide any unique requirements. The format of the proposed BFN ISTS does not include providing "cross references."
- A9 Surveillance Requirements for MOV operability, and check valves that are required by the Inservice Testing (IST) Program, have been removed from individual Specifications. This change is considered administrative in nature since these requirements remain in the IST Program which is defined by proposed Specification 5.5.6.

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING**

- A10 The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow is tested at both the higher and lower operating ranges of the system. Since the reactor steam dome pressure must be ≥ 920 psig to perform SR 3.5.1.7 and ≥ 150 psig to perform SR 3.5.1.8, sufficient time is allowed after adequate pressure is achieved to perform these tests. This is clarified by a Note in both SRs that state the Surveillances are not required to be performed until 12 hours after the specified reactor steam dome pressure is reached. CTS 3.5.E.1 already contains the context of the Note for the low pressure flow rate test. This is also consistent with interpretation of the current technical specification requirement for the high pressure flow rate test which is currently not modified by a Note.
- A11 The existing Applicabilities for High Pressure Coolant Injection (HPCI) Operability (3.5.E.1) and ADS (3.5.G.1) require the systems to be Operable whenever irradiated fuel is in the vessel and reactor pressure is greater than 150 psig (105 psig for ADS), except in the COLD SHUTDOWN CONDITION. The proposed change (LCO 3.5.1 Applicability) requires HPCI and ADS to be Operable in Modes 1, 2 and 3, except when reactor steam dome pressure is < 150 psig. (Reference Justification L5 for the change in applicability from < 105 psig to < 150 psig for ADS.) This change more clearly defines the conditions when HPCI and ADS are required to be Operable without changing the specific requirements which are currently located in the individual specifications. This change is administrative because the same requirements for Operability currently listed in the specific specifications will be labeled APPLICABILITY and apply to the entire ISTS Section 3.5.1, ECCS-Operating. The 3.5.E.2, 3.5.G.2, and 3.5.G.3 Applicabilities are only cross references and have been deleted.
- A12 A finite Completion Time has been provided to verify RCIC OPERABILITY. The new time is immediately and is considered administrative since this is an acceptable interpretation of the time to perform the current requirement.
- A13 CTS 3.9.A.3.h (for Unit 1 and 2) and 3.9.A.3.g (for Unit 3) require 480 V reactor motor operated valve (RMOV) boards to be energized with motor-generator (MG) sets in service. CTS 3.9.B.13 and 14 (for Unit 1 and 2) and 11 and 12 (for Unit 3) provide Required Actions for when one or any two 480-V MG board sets become inoperable. There are two 480-V AC RMOV boards that contain MG sets in their feeder lines. The 480-V AC RMOV boards provide motive power to valves associated with the LPCI mode of the RHR system. The MG sets act as electrical isolators to prevent a



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING**

fault propagating between electrical divisions due to an automatic transfer. Having an MG set out of service reduces the assurance that full RHR (LPCI) capacity will be available when required, therefore, the unit can only operate in this condition for 7 days. Having two MG sets out of service can considerably reduce equipment availability; therefore, the unit must be placed in Cold Shutdown within 24 hours. The inability to provide power to the inboard injection valve and the recirculation pump discharge valve from either 4 kV board associated with an inoperable MG set would result in declaring the associated LPCI subsystems inoperable and entering the Actions required for LPCI. Since, the out of service times for LPCI and the MG sets are comparable, the deletion of the MG set actions is considered administrative.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 Proposed Action H requires LCO 3.0.3 be entered immediately which requires the plant to be in MODE 2 in 7 hours and MODE 3 within 13 hours when multiple ECCS subsystems are inoperable. This change is more restrictive because it stipulates that the reactor shutdown be completed much earlier than would be required by the existing specifications (CTS 3.5.A.3, 3.5.B.4, 3.5.B.8, and 3.5.E.3). For CTS 3.5.G.2 it is slightly more restrictive since it requires the plant to be in MODE 2 in 7 hours where no action was required before. CTS require a shutdown to MODE 4 within 24 hours (except CTS 3.5.G.2 for ADS which also requires the plant be in MODE 3 in 12 hours) but does not stipulate how quickly MODE 3 must be reached. Reference Comment L12 which addresses the less restrictive change of being in MODE 3 in 13 hours versus 12 hours and MODE 4 (or < 150 psig which is outside the applicability for ADS and HPCI) in 37 hours rather than 24 hours.
- M2 Surveillance requirement SR 3.5.1.3 has been added to verify that ADS air supply header pressure is ≥ 90 psig. This is a new Surveillance Requirement which verifies that sufficient air pressure exists in the ADS accumulators/receivers for reliable operation of ADS. Since this is a new Surveillance Requirement, it is an added restriction to plant operations.
- M3 With the reactor pressure < 105 psig, CTS 3.5.B.2 allows the RHR System to be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING**

that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE. This appears to be an exception to CTS 3.5.A.2 & 3, which only allows one CSS loop (i.e., one loop with two pumps) to be inoperable for 7 days and an immediate shutdown if this cannot be met. The # Note for 3.5.B.1 allows LPCI to be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure < 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable. Proposed Specification 3.5.1 has a similar provision (Note to SR 3.5.1.2). Since the proposed Specification has no provision that would allow continued operation in MODE 3 with pressure <105 psig with two CS loops with one pump per loop OPERABLE, the proposed change is considered more restrictive.

- M4 An additional requirement is being added that requires the plant to be in MODE 3 within 12 hours. This change is more restrictive because it stipulates that the reactor shutdown be completed much earlier than would be required by the existing specifications (CTS 3.5.A.3, 3.5.B.4, 3.5.B.8, and 3.5.E.3). CTS require a shutdown to MODE 4 within 24 hours but does not stipulate how quickly MODE 3 must be reached. Reference Comment L2 which addresses the less restrictive change of being in MODE 4 (or < 150 psig for HPCI and ADS) in 36 hours rather than 24 hours.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA1 Not used.

LA2 The details relating to system design and purpose have been relocated to the Bases. The design features and system operation are also described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the FSAR will be controlled by the provisions of 10 CFR 50.59. ECCS system operability determinations are described in the Bases. SR 3.5.1.1 will ensure maintenance of filled discharge piping.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING

- LA3 Details of the methods of performing surveillance test requirements and routine system status monitoring have been relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.
- LA4 Any time the OPERABILITY of a system or component has been affected by repair, maintenance or replacement of a component, post maintenance testing is required to demonstrate OPERABILITY of the system or component. Therefore, explicit post maintenance Surveillance Requirements have been deleted from the Specifications. Also, proposed SR 3.0.1 and SR 3.0.4 require Surveillances to be current prior to declaring components operable.
- LA5 CTS 3.5.D/4.5.D, Equipment Area Coolers, are being relocated to plant procedures. Relocating requirements for the equipment area coolers does not preclude them from being maintained operable. They are required to be operable in order to support HPCI, RCIC, LPCI and CS system operability. If they become inoperable, the operability of the supported systems are required to be evaluated under the Safety Function Determination Program in Section 5.0 of the Technical Specifications. This change is consistent with NUREG-1433.
- LA6 CTS 3.5.E specifically states that HPCI Operability can be determined prior to startup by using an auxiliary steam supply in lieu of using reactor steam after reactor steam dome pressure reaches 150 psig. Details of the methods of performing this surveillance test requirement have been relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.
- LA7 CTS 4.5.H.1 requires the discharge piping of RHR (LPCI and Containment Spray) to be vented from the high point and water level determined every month and prior to testing of these systems. The specific requirement to vent prior to testing has been relocated to procedures. Changes to the procedures will be controlled by the licensee controlled programs.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING

"Specific"

- L1 The phrase "actual or," in reference to the automatic initiation signal, has been added to the surveillance requirement for verifying the ECCS subsystems/ADS actuate on an automatic initiation signal. This allows satisfactory automatic system initiations for other than surveillance purposes to be used to fulfill this requirement. Operability is adequately demonstrated in either case since the ECCS subsystems/ADS itself can not discriminate between "actual" or "simulated."
- L2 The time to reach MODE 4, Cold Shutdown (for LPCI and CS) and < 150 psig (for HPCI and ADS) has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added for LPCI, CS and HPCI (Reference Comment M4 above). These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.
- L3 A new Action (proposed ACTION D) is being added to LCO 3.5.1 for the condition of an inoperable HPCI System coincident with one inoperable low pressure ECCS injection/spray subsystem. The analysis summarized in the current SAFER/GESTR-LOCA analysis (NEDC-32484P, February 1996) demonstrates that adequate cooling is provided by the ADS system and the remaining operable low pressure injection/spray subsystems. However, the redundancy has been reduced such that another single failure may not maintain the ability to provide adequate core cooling. Therefore, an allowable outage time of 72 hours has been assigned to restore either the inoperable HPCI system or the inoperable low pressure injection/spray subsystem to operability. This change is consistent with NUREG-1433.
- L4 The allowable outage time for HPCI has been extended from 7 days to 14 days. Adequate core cooling can be provided by ADS and the low pressure ECCS subsystems. The 14 days is allowed only if all six ADS valves and the low pressure ECCS subsystems are operable. (The exception, LCO 3.5.1, Condition D, which allows operation for 72 hours with HPCI and one low pressure ECCS subsystem inoperable is addressed in Comment L3 above.) The 14 day Completion Time is based on the reliability study that evaluated the impact on ECCS availability (Memorandum from R. L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING**

LCOs for ECCS Components," December 1, 1975). Factors contributing to the acceptability of allowing continued operations for 14 days with HPCI inoperable include: the similar functions of HPCI and RCIC, and that the RCIC is capable of performing the HPCI function, although at a substantially lower capacity; the continued availability of the full complement of ADS valves and the ADS System's capability in response to a small break LOCA; and, the continued availability of the full complement of low pressure ECCS subsystems which, in conjunction with ADS, are capable of responding to a small break LOCA. This change is consistent with NUREG-1433.

- L5 The pressure at which ADS is required to be operable is increased to 150 psig to provide consistency of the operability requirements for HPCI and RCIC equipment. Small break loss of coolant accidents are not analyzed to occur at low pressures (i.e., between 105 and 150 psig). The ADS is required to operate to lower the pressure sufficiently so that the LPCI and CS systems can provide makeup to mitigate such accidents. Since these systems can begin to inject water into the reactor pressure vessel at pressures well above 150 psig, there is no safety significance in the ADS not being operable between 105 and 150 psig.
- L6 A new ACTION has been added (ACTION F), which allows an outage time of 72 hours when one ADS valve and a low pressure ECCS subsystem is inoperable. Currently, there is no allowed outage time when these two items are inoperable. The analysis summarized in the current SAFER/GESTR-LOCA analysis (NEDC-32484P, February 1996) demonstrates that adequate cooling is provided by the HPCI and the remaining operable low pressure injection/spray system. However, the redundancy has been reduced such that another single failure concurrent with a design basis LOCA could result in the minimum required ECCS equipment not being available. Therefore, an allowable outage time of 72 hours has been assigned to restore either the inoperable ADS valve or the inoperable low pressure injection/spray system. This change is consistent with NUREG-1433.
- L7 Current Technical Specifications only allow one LPCI pump to be inoperable. Proposed ACTION A allows two LPCI pumps, one per loop or two in one loop, to be inoperable for seven days. The BASES for ISTS 3.5.1 Required Action A.1 state that the 7 day allowed outage time is justified because in this condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. This justification is applicable for the LPCI function of RHR with one or two RHR (LPCI) pumps out of service as demonstrated by previous LOCA analyses performed for



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING

BFN as well as the current SAFER/GESTR-LOCA analysis (NEDC-32484P, February 1996). Following postulated single failures, adequate core cooling can be provided by one loop of Core Spray (2 pumps) and two RHR (LPCI) pumps (either two pumps in one loop or one pump in two loops) in conjunction with HPCI and ADS. Therefore, this less restrictive change is acceptable based on the plant specific LOCA analysis performed for BFN.

- L8 This change proposes to add a Note to current Surveillance Requirement 4.6.D.4 (proposed Surveillance Requirement 3.5.1.12) which states, "Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test." This change allows the Applicability of the Specification to be entered for 12 hours without performing the Surveillance Requirement. This allows for sufficient conditions to exist and allow the plant to stabilize within these conditions prior to performing the Surveillance. The normal outcome of the performance of a Surveillance is the successful completion which proves Operability. This change represents a relaxation over existing requirements. This change is consistent with NUREG-1433.
- L9 Existing Surveillance Requirement 4.5.E.1.d requires verification that HPCI is capable of delivering at least 5000 gpm at normal reactor vessel operating pressure. The proposed surveillance, SR 3.5.1.7, requires verification of a minimum 5000 gpm HPCI flow rate with reactor pressure ≥ 920 psig and < 1010 psig. The HPCI performance test at high pressure is the second part of a two part test that verifies HPCI pump performance at the upper and lower end of the range of steam supply and pump discharge pressures in which the HPCI pump is expected to perform. Performance of the HPCI test at both ends of the expected operating pressure range confirms that the HPCI pump and turbine are functioning in accordance with design specifications. The ability of the HPCI pump to perform at normal reactor vessel operating pressure has already been demonstrated. A small decrease in the pressure to as low as 920 psig at which the performance to design specifications is verified will not affect the validity of the test to determine that the pump and turbine are still operating at the design specifications.



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING**

L10 Existing Surveillance Requirement 4.5.C.1.e requires verification that HPCI is capable of delivering at least 5000 gpm "at 150 psig reactor steam pressure." The proposed surveillance, SR 3.5.1.9, requires verification of a minimum 5000 gpm HPCI flow rate with reactor pressure at \leq 165 psig. This change is less restrictive because it could allow reactor operation at pressures up to 165 psig prior to performing the surveillance. Performance of HPCI pump testing draws steam from the reactor and could affect reactor pressure significantly. Therefore, HPCI pump testing must be performed when the Electro-Hydraulic Control (EHC) System for the main turbine is available and capable of regulating reactor pressure. Operating experience has demonstrated that reactor pressures as high as 165 psig may be required before the EHC system is capable of maintaining stable pressure during the performance of the HPCI test.

The HPCI performance test at low pressure is the first part of a two part test that verifies HPCI pump performance at the upper and lower end of the range of steam supply and pump discharge pressures in which the HPCI pump is expected to perform. Performance of the HPCI test at both ends of the expected operating pressure range confirms that the HPCI pump and turbine are functioning in accordance with design specifications. The ability of the HPCI pump to perform at the lowest required pressure of 150 psig has already been demonstrated. A small increase in the pressure at which the performance to design specifications is verified will not significantly delay or affect the validity of the test to determine that the pump and turbine are still operating at the design specifications.

L11 CTS 3.5.E.1 requires HPCI operability to be determined within 12 hours after reactor steam dome pressure reaches 150 psig from a COLD CONDITION. The proposed Note to SR 3.5.1.7 and 3.5.1.8 allows 12 hours to perform the test after reactor steam dome pressure and flow are adequate. This is based on the need to reach conditions appropriate for testing. The existing allowance to reach a given pressure only partially addresses the issue. This pressure can be attained, and with little or no steam flow, conditions would not be adequate to perform the test - potentially resulting in an undesired reactor depressurization. The proposed change recognizes the necessary conditions of steam flow and minimum pressure as well as a maximum pressure limitation and provides consistency of presentation of these conditions. The point in time during startup that testing would begin remains unchanged. The change simply changes when the 12 hour clock for performing the test

JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING

must begin and permits testing to be completed in a reasonable period of time.

- L12 Proposed Condition H provides direction for various interrelationships between HPCI and ADS, and Between LPCI and CS. The Action requires entry into LCO 3.0.3 for various combinations of inoperability which are consistent with the present required actions for the same various combinations (CTS 3.5.A.3, 3.5.B.4, 3.5.B.8, and 3.5.E.3). However, the time to reach MODE 4, Cold Shutdown (for LPCI and CS) and < 150 psig (for HPCI) has been extended from 24 hours to 37 hours and to reach MODE 3, Hot Shutdown (for ADS only) has been extended from 12 hours to 13 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 2 in 7 hours and MODE 3 (Hot Shutdown) within 13 hours has been added (Reference Comment M1 above). These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.
- L13 An alternate verification to ensure the LPCI cross tie between loops is isolated has been added for Unit 3. The addition of an alternate method of satisfying the surveillance requirement is considered less restrictive. Currently, the method used for all three units is to verify the LPCI cross tie is closed and power is removed from the valve operator. Unit 3 has a manual shutoff valve install between the cross tie for Loop I and Loop II. This verification ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other subsystem. Since the manual shutoff valve serves the same function as the power operated valve, the proposed change is considered acceptable.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING

RELOCATED SPECIFICATIONS

- RI. Browns Ferry Nuclear Plant consists of three units. The pump suction and heat exchanger discharge lines of one loop of RHR in Unit 1 (Loop II) are cross-connected to the pump suction and heat exchanger of Unit 2. Unit 2 and 3 systems are cross-connected in a similar manner. Technical Specification requirements related to RHR cross-tie capability between units have been deleted. The standby coolant supply connection and RHR crossties are provided to maintain long-term reactor core and primary containment cooling capability irrespective of primary containment integrity or operability of the RHR System associated with a given unit. They provide added long-term redundancy to the other ECC Systems and are designed to accommodate certain situations which, although unlikely to occur, could jeopardize the functioning of these systems. Neither the RHR cross-tie nor the standby coolant supply capability is assumed to function for mitigation of any transient or accident analyzed in the FSAR. Therefore, the operability requirements and surveillances associated with the cross-connection capability have been relocated to the Technical Requirements Manual (TRM). Changes to the TRM will be controlled in accordance with 10 CFR 50.59.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

AUG 02 1989

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.5.A Core Spray System (CSS)

See Justification for Changes for BFN ISTS 3.5.1

- 2. If one CSS loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are OPERABLE.
- 3. If Specification 3.5.A.1 or Specification 3.5.A.2 cannot be met, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

Applicability {

When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one OPERABLE pump and associated diesel generator shall be OPERABLE, except with the reactor vessel head removed as specified in 3.5.A.5 or PRIOR TO STARTUP as specified in 3.5.A.1.

LCO 3.5.2 L2

See Justification for Changes for BFN ISTS 3.5.1

4.5.A Core Spray System (CSS)

4.5.A.1.d (Cont'd)

105 psi differential pressure between the reactor vessel and the primary containment.

- e. Check Valve Per Specification 1.0.MM
- f. Verify that Once/Month each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct* position.
- 2. No additional surveillance is required.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

See Justification for Changes for BFN ISTS 3.5.2



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.A Core Spray System (CSS)~~

* 5. When irradiated fuel is in the reactor vessel and the reactor vessel head is removed, core spray is not required to be OPERABLE provided the cavity is flooded, the fuel pool gates are open and the fuel pool water level is maintained above the low level alarm point, and provided one ~~RHR~~ pump and associated valves supplying the standby coolant supply are OPERABLE.

LCO Applicability

(R1)

(M1) Proposed ACTIONS →

* When work is in progress which has the potential to drain the vessel, manual initiation capability of either 1 CSS Loop or 1 RHR pump, with the capability of injecting water into the reactor vessel, and the associated diesel generator(s) are required.

(L1)

See Justification for Changes for BFN ISTS 3.8.2

(M3)

Proposed SR 3.5.2.4

(M4)

Proposed SR 3.5.2.5 for CSS →

(A1)

~~3.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

8. No additional surveillance required.

See Justification for Changes for BFN 15TS 3.5.1

SR3.5.2.5

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be (A2) demonstrated to be OPERABLE per Specification 1.0.MM.

See Justification for Changes for BFN 15TS 3.8.2

Applicability

LCO 3.5.2 L2

Note for SR 3.5.2.4

LCO Applicability

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

~~10. No additional surveillance required.~~

(A1)

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification for Changes for BFN 15TS 3.5.1



MAY 19 1994

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.H. Maintenance of Filled Discharge Pipe~~

~~4.5.H. Maintenance of Filled Discharge Pipe~~

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

See Justification for Changes for BFN ISTS 3.5.1 + 3.5.3

SR 3.5.2.3

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

1. (LA1) Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined. (LA3)
2. (LA2) Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.
3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.
4. (LA3) When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.



3.7/4.7 CONTAINMENT SYSTEMS

See Justification
Changes for BFN
ISTS 3.6.2.1+2

Specification 3.5.2

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

a. Minimum water level =
-6.25" (differential pressure control >0 psid)
-7.25" (0 psid differential pressure control)

b. Maximum water level =
-1"

SR 3.5.2.1

Add Applic of LCO 3.5.2

M2

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

~~A. Primary Containment~~

~~1. Pressure Suppression Chamber~~

SR 3.5.2.1

a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

M2

12 hrs



MAR 09 1994

~~3.9/A.9 AUXILIARY ELECTRICAL SYSTEM~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.9.C. Operation in Cold Shutdown

Whenever the reactor is in COLD SHUTDOWN CONDITION with irradiated fuel in the reactor, the availability of electric power shall be as specified in Section 3.9.A except as specified herein.

- 1. At least two units 1 and 2 diesel generators and their associated 4-kV shutdown boards shall be OPERABLE.
- 2. An additional source of power energized and capable of supplying power to the units 1 and 2 shutdown boards consisting of at least one of the following:
 - a. One of the offsite power sources specified in 3.9.A.1.c.
 - b. A third OPERABLE diesel generator.
- 3. At least one 480-V shutdown board for each unit must be OPERABLE.

- 4. One 480-V RMOV board mg set is required for each RMOV board (1D or 1E) required to support operation of the RHR system in accordance with 3.5.B.9.

4.9.C. Operation in Cold Shutdown

- 1. No additional surveillance is required.

See Justification for Changes for BFN ISTS Section 3.8

(A3)



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



AUG 02 1989

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.5.A Core Spray System (CSS)

4.5.A Core Spray System (CSS)

4.5.A.1.d (Cont'd)

105 psi differential pressure between the reactor vessel and the primary containment.

- e. Check Valve Per Specification 1.0.MM
- f. Verify that Once/Month each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct* position.
- 2. No additional surveillance is required.

See Justification for Changes for BFN 1STS 3.5.1

- 2. If one CSS loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are OPERABLE.
- 3. If Specification 3.5.A.1 or Specification 3.5.A.2 cannot be met, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one OPERABLE pump and associated diesel generator shall be OPERABLE, except with the reactor vessel head removed as specified in 3.5.A.5 or PRIOR TO STARTUP as specified in 3.5.A.1.

Applicability LCO 3.5.2 L2

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

See Justification for Changes for BFN 1STS 3.5.1

See Justification for Changes for BFN 1STS 3.8.2



3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

DEC 15 1988

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.5.A ~~Core Spray System (CSS)~~

LCO 3.5.2
Applicability

- * 5. When irradiated fuel is in the reactor vessel and the reactor vessel head is removed, core spray is not required to be OPERABLE provided the cavity is flooded, the fuel pool gates are open and the fuel pool water level is maintained above the low level alarm point, and provided one RHRSW pump and associated valves supplying the standby coolant supply are OPERABLE.

(RI)

(MI) Proposed ACTIONS.

* When work is in progress which has the potential to drain the vessel, manual initiation capability of either 1 CSS Loop or 1 RHR pump, with the capability of injecting water into the reactor vessel, and the associated diesel generator(s) are required.

See Justification for Changes for BFN ISTS 3.8.2

(M3)
Proposed SR 3.5.2.4

(M4)
Proposed SR 3.5.2.5 for CSS

(LI)



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

AI

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

8. No additional surveillance required.

See Justification for Changes for BFN ISTS 3.5.1

9. (When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.)

Applicability

LCO 3.5.2

L2

Note for SR 3.5.2.4

SR 3.5.2.5

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

See Justification for Changes for BFN ISTS 3.8.2

LCO Applicability

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

AI

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification for Changes for BFN ISTS 3.5.1



AUG 02 1989

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~~~LIMITING CONDITIONS FOR OPERATION~~

A1

~~SURVEILLANCE REQUIREMENTS~~~~3.5.H. Maintenance of Filled Discharge Pipe~~

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

See Justification for Change
for BFN ISTS 3.5.1 & 3.5.3

~~4.5.H. Maintenance of Filled Discharge Pipe~~~~SR 3.5.2.3~~

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined. (LA1)
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service. (LA2)
3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.
4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded. (LA3)



See Justification for
Changes for BFN
1573 3.6.2.1+2

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

a. Minimum water level =
-6.25" (differential pressure control >0 psid)
-7.25" (0 psid differential pressure control)

b. Maximum water level =
-1"

SRs
3.5.2.1

Add applic of LCO 3.5.2

M2

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

1. Pressure Suppression Chamber

SR
3.5.2.1

a. The suppression chamber water level be checked once per ^{12hrs} day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

M2

JAN 09 1991

3.9/4.9 AUXILIARY ELECTRICAL SYSTEM

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.9.C. Operation in Cold Shutdown

Whenever the reactor is in COLD SHUTDOWN CONDITION with irradiated fuel in the reactor, the availability of electric power shall be as specified in Section 3.9.A except as specified herein.

1. At least two Units 1 and 2 diesel generators and their associated 4-kV shutdown boards shall be OPERABLE.
2. An additional source of power energized and capable of supplying power to the Units 1 and 2 shutdown boards consisting of at least one of the following:
 - a. One of the offsite power sources specified in 3.9.A.1.c.
 - b. A third OPERABLE diesel generator.
3. At least one 480-V shutdown board for each unit must be OPERABLE.

4. One 480-V RMOV board mg set is required for each RMOV board (2D or 2E) required to support operation of the RHR system in accordance with 3.5.B.9.

4.9.C Operation in Cold Shutdown

- i. No additional surveillance is required.

See Justification for Changes for BFN ISTS Section 3.8

(A3)



UNIT 3
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

3.5.A Core Spray System (CSS)

4.5.A Core Spray System (CSS)

4.5.A.1.d (Cont'd)

105 psi differential pressure between the reactor vessel and the primary containment.

e. Testable Check Valve Per Specification I.O.MM

f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month

2. No additional surveillance is required.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

- 2. If one CSS loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are OPERABLE.
- 3. If Specification 3.5.A.1 or Specification 3.5.A.2 cannot be met, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one OPERABLE pump and associated diesel generator shall be OPERABLE, except with the reactor vessel head removed as specified in 3.5.A.5 or PRIOR TO STARTUP as specified in 3.5.A.1.

Applicability {

LCO 3.5.2 {

(L2)

See Justification for Changes for BFN ISTS 3.5.1

See Justification for Changes for ISTS 3.5.1

See Justification for Changes for BFN ISTS 3.5.2



DEC 15 1988

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATIONS~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.A Core Spray System (CSS)~~

* 5. When irradiated fuel is in the reactor vessel and the reactor vessel head is removed, core spray is not required to be OPERABLE provided the cavity is flooded, the fuel pool gates are open and the fuel pool water level is maintained above the low level alarm point, and provided one RHRW pump and associated valves supplying the standby coolant supply are OPERABLE.

LCO 3.5.2 Applicability

(R1)

(M1)

Proposed ACTIONS

* When work is in progress which has the potential to drain the vessel, manual initiation capability of either 1 CSS Loop or 1 RHR pump, with the capability of injecting water into the reactor vessel, and the associated diesel generator(s) are required.

See Justification for Changes for BFN 1STS 3.8.2

(M3)

Proposed SR 3.5.2.4

(M4)

Proposed SR 3.5.2.5 for CSS

(L1)



Specification 3.5.2

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

APR 19 1994

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

8. No additional surveillance required.

See Justification for Changes for BFN 15TS 3.5.1

Applicability

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

SR 3.5.2.5
9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

See Justification for Changes for BFN 15TS 3.8.2

LCO 3.5.2 (2)

Note for SR 3.5.2.4

(A1)

LCO Applicability

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The B and D RHR pumps on unit 2 which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification for Changes for BFN 15TS 3.5.1



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.H Maintenance of Filled Discharge Pipe~~

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

See Justification for Changes for BFN ISTS 3.5.1 & 3.5.3

~~4.5.H Maintenance of Filled Discharge Pipe~~

SR 3.5.2.3

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow determined. (LA1, LA3)
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service. (LA2)
3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.
4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded. (LA3)



3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

SR 3.5.2.1

Add applicability of LCO 3.5.2

M2

- a. Minimum water level =
 - 6.25" (differential pressure control >0 psid)
 - 7.25" (0 psid differential pressure control)

- b. Maximum water level = -1"

See Justification for Changes for BFN ISTS 3.6.2.1 + 2

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

- ~~1. Pressure Suppression Chamber~~

SR 3.5.2.1

M2

12hrs day

- a. The suppression chamber water level be checked once per 12hrs day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.



(A1)

3.9.C. Operation in COLD SHUTDOWN CONDITION

Whenever the reactor is in the COLD SHUTDOWN CONDITION with irradiated fuel in the reactor, the availability of electric power shall be as specified in Section 3.9.A except as specified herein.

1. At least two Unit 3 diesel generators and their associated 4-kV shutdown boards shall be OPERABLE.
2. An additional source of power energized and capable of supplying power to the Unit 3 shutdown boards consisting of at least one of the following:
 - a. One of the offsite power sources specified in 3.9.A.1.c.
 - b. A third OPERABLE diesel generator.
3. At least one Unit 3 480-V shutdown board must be OPERABLE.

~~4. One 480-V RMOV board motor generator (mg) set is required for each RMOV board (3D or 3E) required to support operation of the RHR system in accordance with 3.5.B.9.~~

4.9.C Operation in COLD SHUTDOWN CONDITION

1. No additional surveillance is required.

see Justification for Changes for BFN 15TS Section 3.8

(A3)

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.2 - ECCS - SHUTDOWN**

ADMINISTRATIVE CHANGES

A1¹ Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 Surveillance Requirements for MOV operability that are required by the Inservice Testing (IST) Program have been removed from individual Specifications. This change is considered administrative in nature since these requirements remain in the IST Program which is defined by proposed Specification 5.5.6.

A3 CTS 3.9.C.4 requires one 480 V reactor motor operated valve (RMOV) board motor-generator (MG) set for each RMOV board required to support the RHR System in accordance with CTS 3.5.B.9. The 480-V AC RMOV boards provide motive power to valves associated with the LPCI mode of the RHR system. The MG sets act as electrical isolators to prevent a fault propagating between electrical divisions due to an automatic transfer. The inability to provide power to the inboard injection valve and the recirculation pump discharge valve from either 4 kV board associated with an inoperable MG set would result in declaring the associated LPCI subsystems inoperable and entering the Actions required for LPCI. Therefore, the deletion of the operability requirement associated with the MG sets in CTS 3.9.C.4 is considered administrative.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.2 - ECCS - SHUTDOWN

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 Proposed ACTIONS A, B, C and D have been added to provide required actions be taken when LCO requirements can not be met. CTS 3.5.A.4 and 3.5.B.9 provide minimum requirements for ECCS subsystems when in MODE 4 and 5 (except with the spent fuel pool gates removed and water level \geq the low level alarm setpoint of the spent fuel pool) but no action if these requirements are not met. Therefore, technical specifications are violated when these requirements can not be met and the default to TS 1.0.C.1 requires no action since the plant is already in Cold Shutdown. While from a compliance standpoint the proposed ACTIONS are less restrictive, from an operational perspective they are more restrictive since actions are required where there were none before. Proposed ACTION A allows 4 hours to restore a subsystem when only one of the required subsystems is inoperable and then proposed ACTION B requires action be initiated to suspend operations with a potential for draining the reactor vessel (OPDRVs) immediately. The 4 hour Completion Time is considered acceptable based on engineering judgment that considers the remaining available subsystem and the low probability of a vessel draindown event during this period. With no required ECCS injection spray subsystems inoperable, proposed ACTION C requires action to be initiated immediately to suspend OPDRVs and at least one required subsystem be restored to OPERABLE status within 4 hours. If one subsystem can not be restored within four hours then Proposed ACTION D requires action be initiated immediately to restore secondary containment to OPERABLE status, to restore two standby gas treatment systems to OPERABLE status, and to restore isolation capability in each required secondary containment penetration flow path not isolated. These actions must be immediately initiated to minimize the probability of a vessel draindown and the subsequent potential for fission product release.
- M2 Proposed SR 3.5.2.1 has been added. SR 3.5.2.1 requires the suppression pool water be verified \geq a minimum level every 12 hours. CTS 3.7.A.1 (& 4.7.A.1.a) requires the suppression pool be verified \geq -6.25" with no differential pressure control once per day at any time irradiated fuel is in the reactor vessel, and the nuclear system is pressurized or work is being done which has the potential to drain the vessel. Therefore, proposed SR 3.5.2.1 is more restrictive since the frequency of performance has been increased from once per 24 hours to once per 12 hours. In addition, CTS only requires performance during atmospheric conditions when work is being done that has the potential to drain the vessel. Therefore, the proposed SR is more restrictive since it



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.2 - ECCS - SHUTDOWN

requires performance during MODES 4, and 5, except with the spent fuel storage pool gates removed and water greater than or equal to minimum level over the top of the reactor pressure vessel flange. The CTS requirement to check the maximum level during OPDRVs has not been included since Specification 3.5.2 concerns the ability to maintain reactor water level using the suppression pool as a source of water. However, this level check is required for proposed Specifications 3.6.2.1 and 3.6.2.2 as it relates to Containment Systems.

- M3 Proposed SR 3.5.2.4, which requires a verification every 31 days that ECCS injection/spray valves are in their correct position, has been added. This provides assurance that the proper flow paths will exist for ECCS operation. This is more restrictive since BFN currently only requires this check during MODES 1, 2 and 3.
- M4 An SR has been added to require a system flow rate test for the Core Spray System during atmospheric conditions. While CTS (4.5.B.9) requires flow rate testing of the RHR pumps during atmospheric conditions as well as during MODES 1, 2, and 3, it only requires CSS flow rate testing during MODES 1, 2, and 3. The addition of this requirement is more restrictive.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

- LA1 CTS 4.5.H.1 requires the discharge piping of RHR (LPCI and Containment Spray) to be vented from the high point and water level determined every month and prior to testing of these systems. The specific requirement to vent prior to testing has been relocated to procedures. Changes to the procedures will be controlled by the licensee controlled programs.
- LA2 Any time the OPERABILITY of a system or component has been affected by repair, maintenance or replacement of a component, post maintenance testing is required to demonstrate OPERABILITY of the system or component. Therefore, explicit post maintenance Surveillance Requirements have been deleted from the Specifications. Also, proposed SR 3.0.1 and SR 3.0.4 require Surveillances to be current prior to declaring components operable.

PAGE 3 OF 5



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.2 - ECCS - SHUTDOWN**

LA3 Details of the methods of performing surveillance test requirements and routine system status monitoring have been relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

L1 CTS 3.5.A.5 requires manual initiation capability of either 1 CSS Loop or 1 RHR pump with capability of injecting water into the reactor vessel when work is in progress which has the potential to drain the vessel. The proposed Specification would not require the CSS or RHR (LPCI and containment cooling mode) system to be operable since LCO 3.5.2 applicability does not apply when the fuel pool gates are open and the fuel pool water level is maintained above the low level alarm setpoint. Therefore, the deletion of this requirement is considered less restrictive. The deletion is acceptable since the coolant inventory represented by this water level is sufficient to allow operator action to terminate the inventory loss prior to fuel uncovering in case of an inadvertent draindown.

L2 The proposed LCO for ECCS-Shutdown is less restrictive since it only requires two low pressure ECCS subsystems to be OPERABLE. This can be fulfilled with any combination of RHR and CS subsystems. That is, two CS subsystems (a CS subsystem for Specification 3.5.2 consists of at least one pump in one loop), two RHR subsystems (RHR subsystem for Specification 3.5.2 consists of one pump in one loop), or one RHR subsystem and one CS subsystem OPERABLE. CTS 3.5.B.9 requires one RHR loop with two pumps or two RHR loops with one pump per loop to be OPERABLE. CTS 3.5.B.4 requires one CS loop with one pump per loop to be OPERABLE. Per CTS 3.5.A Bases the minimum requirement at atmospheric pressure is for one supply of makeup water to the core. Therefore, requiring two RHR pumps and one CS pump to be OPERABLE provides excess redundancy. In addition, since only one supply of makeup water is required, sufficient makeup water can be provided by two CS subsystems, two RHR subsystems, or one CS and one RHR subsystem. As such, the proposed Specification ensures redundancy by requiring any two low pressure ECCS subsystems to be OPERABLE.

PAGE 4 OF 5

JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.2 - ECCS - SHUTDOWN

RELOCATED SPECIFICATIONS

R1 Browns Ferry Nuclear Plant consists of three units. The pump suction and heat exchanger discharge lines of one loop of RHR in Unit 1 (Loop II) are cross-connected to the pump suction and heat exchanger of Unit 2. Unit 2 and 3 systems are cross-connected in a similar manner. Technical Specification requirements related to RHR cross-tie capability between units have been deleted. The standby coolant supply connection and RHR crossties are provided to maintain long-term reactor core and primary containment cooling capability irrespective of primary containment integrity or operability of the RHR System associated with a given unit. They provide added long-term redundancy to the other ECC Systems and are designed to accommodate certain situations which, although unlikely to occur, could jeopardize the functioning of these systems. Neither the RHR cross-tie nor the standby coolant supply capability is assumed to function for mitigation of any transient or accident analyzed in the FSAR. Therefore, the operability requirements and surveillances associated with the cross-connection capability have been relocated to the Technical Requirements Manual (TRM). Relocation to the TRM is in accordance with the "Application of Selection Criteria to BFN TS" and the NRC Final Policy Statement on Technical Specification Improvements. Refer to the application document discussion for additional information.

UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

(A1)

3.5.E High Pressure Coolant Injection System (HPCIS)

See Justification for Changes for BFN ISTS 3.5.1

2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS (LPCI), and RCIGS are OPERABLE.

3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

e. Flow Rate at Once/18 150 psig months

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

f. Verify that Once/Month each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

~~F. Reactor Core Isolation Cooling System (RCIGS)~~

LCO 3.5.3

Applicability

1. The RCIGS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

BFN Unit 1

Proposed Note for SR 3.5.34

3.5/4.5-14

(A1) ~~F. Reactor Core Isolation Cooling System (RCIGS)~~

1. RCIG Subsystem testing shall be performed as follows:

(Actual of L1)

SR 3.5.3.5

a. Simulated Auto- Once/18 matic Actuation months Test

(A3)

Proposed Note for SR 3.5.3.5

AMENDMENT NO. 180



NOV 24 1989

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.F Reactor Core Isolation Cooling System (RCICS)~~

~~4.5.F Reactor Core Isolation Cooling System (RCICS)~~

~~3.5.F.1 (Cont'd)~~

~~4.5.F.1 (Cont'd)~~

Proposed Note for SR 3.5.3.4

be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

SR 3.5.3.3 b. Pump OPERABILITY

Per Specification 1.0.MM

A4 c. Motor-Operated Valve OPERABILITY

Per Specification 1.0.MM

SR 3.5.3.3 d. Flow Rate at normal reactor vessel operating pressure

Once/3 months

SR 3.5.3.4 e. Flow Rate at 150 psig ≤ 165

Once/18 months

SR 3.5.3.3 SR 7.5.7.4 The RCIC pump shall deliver at least 600 gpm during each flow test.

SR 3.5.3.2 f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct position.

Once/31 days

ACTION A

2. If the RCICS is inoperable, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is OPERABLE during such time.

14 L2 verified immediately A7

3. If Specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurized to less than 150 psig within 24 hours.

L3 36 or equal to A8

A1 2. No additional surveillances are required.

A6 * Except that an automatic valve capable of automatic return to its normal position when a signal is present may be in a position for another mode of operation.



MAY 19 1994

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.H. Maintenance of Filled Discharge Pipe~~

~~4.5.H. Maintenance of Filled Discharge Pipe~~

SR 3.5.3.1

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled. (LA1)

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined.

2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

SR 3.5.3.1

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis. (LA2)

4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

See Justification for Changes for BFN 1STS 3.5.1



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



FEB 07 1991

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

3.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

- e. Flow Rate at Once/18 months
150 psig

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

- f. Verify that Once/Month each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

- 2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

See Justification for changes for BFN ISTS 3.5.1

- 2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS(LPCI), and RCIGS are OPERABLE.

- 3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

~~F. Reactor Core Isolation Cooling System (RCIGS)~~

(A1) ~~F. Reactor Core Isolation Cooling System (RCIGS)~~

LCO 35.3

- 1. The RCIGS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

Applicability

BFN Unit 2

Proposed Note for SR 3.5.3.4

- 1. RCIG Subsystem testing shall be performed as follows:

SR 3.5.3.5

- a. ^(Actual on L1) Simulated Automatic Actuation Test Once/18 months

(A2) Proposed Note for SR 3.5.3.5

3.5/4.5-14

AMENDMENT NO. 190



A1

NOV 24 1989

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.F. Reactor Core Isolation Cooling System (RCIGS)

3.5.F.1 (Cont'd)

Proposed Note for SR 3.5.3.4

be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

L6 and flow

L43

ACTION A

2. If the RCIGS is inoperable, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is OPERABLE during such time.

14 22

verified immediately

3. If Specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurized to less than 150 psig within 24 hours.

L3 36

or equal to

A8

4.5.F. Reactor Core Isolation Cooling System (RCIGS)

4.5.F.1 (Cont'd)

SR 3.5.3.3 b. Pump OPERABILITY

Per Specification 1.0.MM

A2

c. Motor-Operated Valve OPERABILITY

Per Specification 1.0.MM

A1

SR 3.5.3.3 d. Flow Rate at normal reactor vessel operating pressure

92 days

Once/3 months

L4

A5 Proposed Note for SR 3.5.3.3

920 to 1010 psig

SR 3.5.3.4 e. Flow Rate at 150 psig

L5 150 psig <= 165

Once/18 months

The RCIG pump shall deliver at least 600 gpm during each flow test.

SR 3.5.3.2 f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct position.

Once/Month

31 days

M1

Be in MODE 3 in 12 hrs

A1 2. No additional surveillances are required.

A6

* Except that an automatic valve capable of automatic return to its normal position when a signal is present may be in a position for another mode of operation.



AUG 02 1989

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.H. Maintenance of Filled Discharge Pipe

(A1)

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

(LA1)

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

See Justification for Changes for BFN ISTS 3.5.1

4.5.H. Maintenance of Filled Discharge Pipe

SR 3.5.3.1

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined.
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

SR 3.5.3.1

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

(LA2)

4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



(A1)

3.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

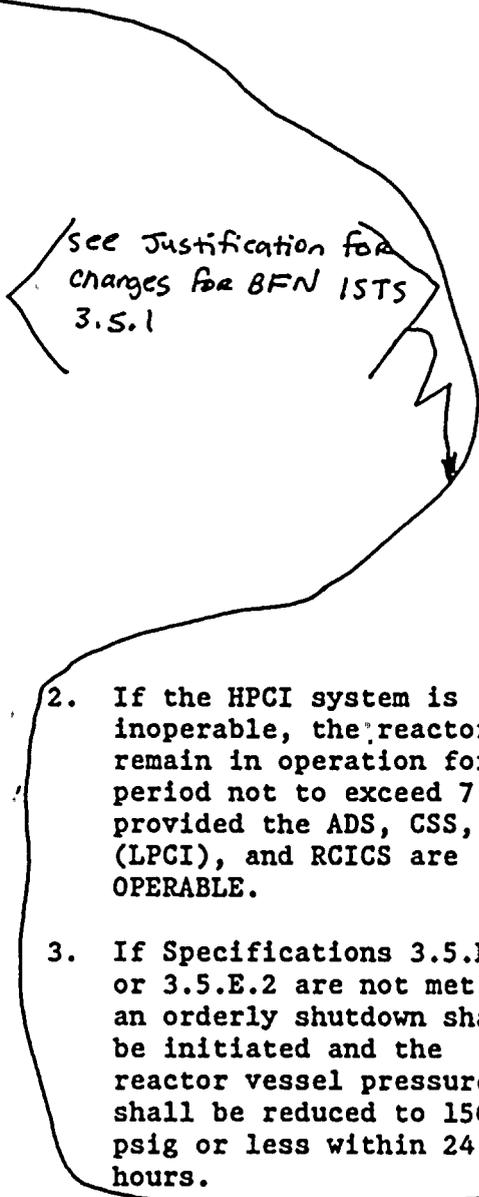
- e. Flow Rate at Once/18 months
150 psig

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

- f. Verify that Once/Month each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

- 2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



- 2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS (LPCI), and RCIGS are OPERABLE.
- 3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

~~F Reactor Core Isolation Cooling System (RCIGS)~~

~~F Reactor Core Isolation Cooling System (RCIGS)~~

LCO 3.5.3

Applicability

- 1. The RCIGS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

(A1)

- 1. RCIG Subsystem testing shall be performed as follows:

Actual or (L1)

- SR 3.5.3.5 a. Simulated Auto- Once/18 matic Actuation months Test

(A3) Proposed Note for SR 3.5.3.5

BFN Unit 3

Proposed Note for SR 3.5.3.4

3.5/4.5-14

AMENDMENT NO. 152



3.5.F Reactor Core Isolation Cooling System (RCIGS)

A1

3.5.F.1 (Cont'd)

L6 and flow

Proposed Note for SR 3.5.3.4

be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

LA3

2. If the RCIGS is inoperable, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is OPERABLE during such time. verified immediately

Action A

14 L2

3. If Specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurized to less than 150 psig within 24 hours.

24 36 L3

DR equal to AB

4.5.F Reactor Core Isolation Cooling System (RCIGS)

4.5.F.1 (Cont'd)

A2

SR 3.5.3.3 b. Pump OPERABILITY

Per Specification 1.0.MM

c. Motor-Operated Valve OPERABILITY

Per Specification 1.0.MM

SR 3.5.3.3 d. Flow Rate at normal reactor vessel operating pressure

Once/3 months

SR 3.5.3.4 e. Flow Rate at 150 psig ≤ 165

Once/18 months

The RCIG pump shall deliver at least 600 gpm during each flow test.

SR 3.5.3.2 f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct position.

Once/Mon. 31 days

2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its normal position when a signal is present may be in a position for another mode of operation.



MAY 19 1994

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

~~3.5.H Maintenance of Filled Discharge Pipe~~

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

(LAI)

The suction of the RCIC (and HPCI) pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

See Justification for Changes for BFN ISTS 3.5.1

~~4.5.H Maintenance of Filled Discharge Pipe~~

SR 3.5.3.1

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow determined.
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

SR 3.5.3.1

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

(LA2)

4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.3 - RCIC SYSTEM

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The Frequency of "Once/month" has been changed to "31 days." The Frequencies of "Once/3 months" and "Per Specification 1.0MM" have been changed to "92 days." Since the proposed frequencies are equivalent, this change is considered administrative.
- A3 Notes allowing actual vessel injection to be excluded from this test (simulated automatic actuation test) have been added to proposed SR 3.5.3.5. Since the current requirements state the test is "simulated" (i.e., valve actuation and vessel injection are inherently excluded), this allowance is considered administrative in nature.
- A4 Surveillance Requirements for MOV operability that are required by the Inservice Testing Program have been removed from individual Specifications. This change is considered administrative in nature since these requirements remain in the IST Program which is defined by proposed Specification 5.5.6.
- A5 The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow is tested at both the higher and lower operating ranges of the system. Since the reactor steam dome pressure must be ≥ 920 psig to perform SR 3.5.3.3 and ≥ 150 psig to perform SR 3.5.3.4, sufficient time is allowed after adequate pressure is achieved to perform these tests. This is clarified by a Note in both SRs that state the Surveillances are not



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.3 - RCIC SYSTEM

required to be performed until 12 hours after the specified reactor steam dome pressure is reached. CTS 3.5.F.1 already contains the context of the Note for the low pressure flow rate test. This is also consistent with interpretation of the current technical specification requirement for the high pressure flow rate test which is currently not modified by a Note.

- A6 The clarifying information contained in the "*" footnote has been moved to the proposed Bases for SR 3.5.3.2. The intent of the surveillance is to assure that the proper flow paths will exist for RCIC System operation. The Bases clarifies that a valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. Moving this clarifying statement to the Bases is considered administrative in nature.
- A7 A finite Completion Time has been provided to verify HPCI OPERABILITY. the new time is immediately and is considered administrative since this is an acceptable interpretation of the time to perform the current requirement.
- A8 CTS 3.5.F.3 requires the reactor to be depressurized to less than 150 psig when CTS 3.5.F.1 and 2 cannot be met, while CTS 3.5.F.1 requires RCIC to be OPERABLE when reactor vessel pressure is above 150 psig. Proposed Required Action B.2 requires the vessel to be depressurized to \leq 150 psig. Since the intent of CTS is the same even though the CTS shutdown statement does not state "equal to," the addition of this requirement is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 An additional requirement is being added that requires the plant to be in MODE 3 within 12 hours. This change is more restrictive because it stipulates that the reactor shutdown be completed much earlier than would be required by the existing specification (CTS 3.5.F.3). CTS require a shutdown to \leq 150 psig within 24 hours but do not stipulate how quickly MODE 3 must be reached. Reference Comment L3 which addresses the less restrictive change of being \leq 150 psig in 36 hours rather than 24 hours.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.3 - RCIC SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 The details relating to system design and purpose have been relocated to the Bases. The design features and system operation are also described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the FSAR will be controlled by the provisions of 10 CFR 50.59. System operability determination, as described in the Bases and SR 3.5.3.1, will ensure maintenance of filled discharge piping.
- LA2 The details relating to methods of performing surveillance test requirements have been relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.
- LA3 CTS 3.5.F.1 specifically states that RCIC Operability can be determined prior to startup by using an auxiliary steam supply in lieu of using reactor steam after reactor steam dome pressure reaches 150 psig. Details of the methods of performing this surveillance test requirement have been relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 The phrase "actual or," in reference to the automatic initiation signal, has been added to the surveillance requirement for verifying that the RCIC System actuates on an automatic initiation signal. This allows satisfactory automatic system initiations for other than surveillance purposes to be used to fulfill the surveillance requirements. Operability is adequately demonstrated in either case since the RCIC System itself can not discriminate between "actual" or "simulated."
- L2 This change proposes to extend the current allowed outage time for the RCIC System from 7 days to 14 days. The 14 days are allowed only if the HPCI System is verified Operable immediately. Loss of the RCIC System will not affect the overall plant capability to provide makeup inventory at high reactor pressure since the HPCI System is the only high pressure system assumed to function during a LOCA. However, the RCIC System is



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.3 - RCIC SYSTEM

the preferred source of makeup for transients and certain abnormal events with no LOCA (RCIC as opposed to HPCI is the preferred source of makeup coolant because of its relatively small capacity, which allows easier control of the RPV water level). The 14 day completion time is also based on a reliability study that evaluated the impact on ECCS availability (Memorandum from R. L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975). Because of similar functions of HPCI and RCIC, and because HPCI is capable of performing the RCIC function, the allowed outage times determined for HPCI can be applied to RCIC. This change is consistent with NUREG-1433.

- L3 The time to reduce reactor steam dome pressure to ≤ 150 psig has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added (See Comment M1 above). These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.
- L4 Existing Surveillance Requirement 4.5.E.1.d requires verification that RCIC is capable of delivering at least 600 gpm at normal reactor vessel operating pressure. The proposed surveillance, SR 3.5.3.3, requires verification of a minimum 600 gpm RCIC flow rate with reactor pressure ≥ 920 psig and < 1010 psig. The RCIC performance test at high pressure is the second part of a two part test that verifies RCIC pump performance at the upper and lower end of the range of steam supply and pump discharge pressures in which the RCIC pump is expected to perform. Performance of the RCIC test at both ends of the expected operating pressure range confirms that the RCIC pump and turbine are functioning in accordance with design specifications. The ability of the RCIC pump to perform at normal reactor vessel operating pressure has already been demonstrated. A small decrease in the pressure to as low as 920 psig at which the performance to design specifications is verified will not affect the validity of the test to determine that the pump and turbine are still operating at the design specifications.
- L5 Existing Surveillance Requirement 4.5.F.1.e requires verification that RCIC is capable of delivering at least 600 gpm "at 150 psig reactor steam pressure." The proposed surveillance, SR 3.5.3.4, requires verification of a minimum 600 gpm RCIC flow rate with reactor pressure



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.3 - RCIC SYSTEM

at 165 psig. This change is less restrictive because it could allow reactor operation at pressures up to 165 psig prior to performing the surveillance. Performance of RCIC pump testing draws steam from the reactor and could affect reactor pressure significantly. Therefore, RCIC pump testing must be performed when the Electro-Hydraulic Control (EHC) System for the main turbine is available and capable of regulating reactor pressure. Operating experience has demonstrated that reactor pressures as high as 165 psig may be required before the EHC system is capable of maintaining stable pressure during the performance of the RCIC test.

The RCIC performance test at low pressure is the first part of a two part test that verifies RCIC pump performance at the upper and lower end of the range of steam supply and pump discharge pressures in which the RCIC pump is expected to perform. Performance of the RCIC test at both ends of the expected operating pressure range confirms that the RCIC pump and turbine are functioning in accordance with design specifications. The ability of the RCIC pump to perform at the lowest required pressure of 150 psig has already been demonstrated. A small increase in the pressure at which the performance to design specifications is verified will not significantly delay or affect the validity of the test to determine that the pump and turbine are still operating at the design specifications.

- L6 CTS 3.5.F.1 requires operability to be determined within 12 hours after reactor steam dome pressure reaches 150 psig from a COLD CONDITION. The allowance for reactor steam dome pressure and flow to be adequate is based on the need to reach conditions appropriate for testing. The existing allowance to reach a given pressure only partially addresses the issue. This pressure can be attained, and with little or no steam flow, conditions would not be adequate to perform the test - potentially resulting in an undesired reactor depressurization. The proposed change recognizes the necessary conditions of steam flow and minimum pressure as well as a maximum pressure limitation and provides consistency of presentation of these conditions. The point in time during startup that testing would begin remains unchanged. The change simply changes when the 12 hour clock for performing the test must begin and permits testing to be completed in a reasonable period of time.



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5 - ECCS AND RCIC SYSTEM BASES**

The Bases of the current Technical Specifications for this section (3.5.A, B, E, F, G, H, and 4.5) have been completely replaced by revised Bases that reflect the format and applicable content of proposed BFN-UNIT 1, 2, and 3 ISTS Section 3.5, consistent with NUREG-1433. The revised Bases are as shown in the proposed BFN-UNIT 1, 2, and 3 Bases.

UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A. Primary Containment~~

(A2)

~~4.7.A. Primary Containment~~

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel ~~except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).~~

(OPERATE)

LCD 3.6.1.1 Applicability

(M1)

performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

(A2)

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

(A4)

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations), exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours

(LA1)

ACTION A { or the reactor shall be placed in Hot Shutdown
ACTION B { within the next 16 hours

(1)

(12)

(M2)

and cold shutdown in 36 hours

~~2. Integrated Leak Rate Testing~~

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

(LA1)

~~SR 3.6.1.1.1~~

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

~~3.7/4.7 CONTAINMENT SYSTEMS~~

FEB 22 1996

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~A.7.A. Primary Containment~~ c

~~A.7.A.2. (Cont'd)~~ c

(A1)

~~b. Deleted~~ c

~~c. Deleted~~ c

+
+

FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~ ←

~~4.7.A.2. (Cont'd)~~ ←

(A1)

~~d. Deleted~~ ←

+

~~e. Deleted~~ ←

+

~~f. Deleted~~ ←

+



(A1)

~~4.7.A. Primary Containment c~~

~~4.7.A.2. (Cont'd) c~~

SR 3.3.6.1.1

- g. Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN 15TS 3.6.1.2

Note: An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq (0.05 L_a)$ when tested at $\geq Pa$. (2) For door seal leakage, the overall air lock leakage rate is $\leq (0.02 L_a)$ when the air lock is pressurized to $(\geq 2.5 psig)$ for at least 15 minutes).

~~3.7.4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(AG) →

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized, until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

ACTION A →

(1) (M2)

ACTION B →

in MODE 3 in 12 hours + MODE 4 in 36 hours

(M2)

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

See Justification for Changes for BFN ISTS 3.6.1.3

~~3.7/4.7 CONTAINMENT SYSTEMS~~

(A1)

~~INERTING CONNECTIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

J. Continuous Leak Rate Monitor

(LC1)

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

k. Drywell and Torus Surfaces

(A3)

The interior surfaces of the drywell and torus above the level one foot below the normal water line and outside surfaces of the torus below the water line shall be visually inspected each operating cycle for deterioration and any signs of structural damage with particular attention to piping connections and supports and for signs of distress or displacement.

~~3.7.4.7 CONTAINMENT SYSTEMS~~
~~OPERATING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

~~4.7.A Primary Containment~~

3.7.A.4 (Cont'd)

4.7.A.4 (Cont'd)

- c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.
- d. If Specifications 3.7.A.4.a, 3.7.A.4.b, or 3.7.A.4.c cannot be met, the unit shall be placed in a COLD SHUTDOWN CONDITION in an orderly manner within 24 hours.

- c. Each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.2.2.

- SR 3.6.1.1.2
- d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

See Justification for Changes for BFN 15TS 3.6.1.7

(A1)

(AS)

18 mos.

5. Oxygen Concentration

5. Oxygen Concentration

- a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.
- b. Within the 24-hour period subsequent to placing the reactor in the RUN MODE following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.
- c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a COLD SHUTDOWN CONDITION within 24 hours.
- d. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN CONDITION within 24 hours.

- a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.
- b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.
- c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

See Justification for Changes for BFN 15TS 3.6.3.2



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A. Primary Containment~~

~~4.7.A. Primary Containment~~

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

LCD 3.6.1.1
Applicability
M1

(A1)
(A2)
OPERABLE

performing "open vessel" physics tests at power levels not to exceed 5 MW(t).
(A2)

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

(A4)

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours

(LAI)

ACTION A { for the reactor shall be placed in Hot Shutdown
ACTION B { within the next 12 hours
1
M2

2. Integrated Leak Rate Testing
Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

(LAI)

SR 3.6.1.1.1
Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

And Cold Shutdown in 36 hours



~~3.7/4.7 CONTAINMENT SYSTEMS~~

(A1)

Specification 3.6.1.1

FEB 22 1996

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

~~b. Deleted~~

+

~~c. Deleted~~

+

FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

- (A1) ~~d. Deleted~~ †
- ~~e. Deleted~~ †
- ~~f. Deleted~~ †



AI

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

SR3 3.6.1.1

8. Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN ISTS 3.6.1.2

Note: An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq (0.05 L_a)$ when tested at \geq Pa. (2) For door seal leakage, the overall air lock leakage rate is $\leq (0.02 L_a)$ when the air lock is pressurized to ≥ 2.5 psig for at least 15 minutes).



LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h: (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(A6) →

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

ACTION A →

(M2) →

ACTION B →

(M2)

in MODE 3 in 12 hours
+ MODE 4 in 36 hours

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

See Justification for Changes for BFN ISTS 3.6.1.3

(A1)

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

~~j. Continuous Leak Rate Monitor~~

(LC1)

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

~~k. Drywell and Torus Surfaces~~

(A3)

The interior surfaces of the drywell and torus above the level one foot below the normal water line and outside surfaces of the torus below the water line shall be visually inspected each operating cycle for deterioration and any signs of structural damage with particular attention to piping connections and supports and for signs of distress or displacement.



~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(AI)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

~~4.7.A Primary Containment~~

3.7.A.4 (Cont'd)

4.7.A.4 (Cont'd)

c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.

c. Each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.MM.

d. If Specifications 3.7.A.4.a, .b, or .c cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.

SR 3.6.1.1.2
d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

See Justification for changes for BFN ISTS 3.6.1.7

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.3.2

5. Oxygen Concentration

5. Oxygen Concentration

a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100/psig, except as specified in 3.7.A.5.b.

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a Cold Shutdown condition within 24 hours.

c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

d. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.



UNIT 3
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

FEB 22 1996

~~3.7/A.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

A1

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A. Primary Containment~~

A2

2.a. Primary containment integrity shall be **OPERABLE** maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

LCD 3.6.1.1
Applicability

M1

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

A4

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours

LAI

ACTION A { or the reactor shall be placed in Hot Shutdown
ACTION B { within the next 16 hours.

12

M2

and cold shutdown in 36 hours

~~4.7.A. Primary Containment~~

~~2. Integrated Leak Rate Testing~~

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.c shall be taken.

LAI

SR 3.6.1.1.1
Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

~~3.7/4.7 CONTAINMENT SYSTEMS~~ e

Specification 3.6.1.1

(A1)

FEB 22 1996

~~LIMITING CONDITIONS FOR OPERATION~~ e

~~SURVEILLANCE REQUIREMENTS~~ e

~~4.7.A. Primary Containment~~ e

~~4.7.A.2. (Cont'd)~~ e

(A1)

~~b. Deleted~~ e

+

~~c. Deleted~~ e

+

~~3.7/4.7 CONTAINMENT SYSTEMS~~ e

(A1)

Specification 3.6.1.1

FEB 22 1996

~~LIMITING CONDITIONS FOR OPERATION~~ e

~~SURVEILLANCE REQUIREMENTS~~ e

~~4.7.A. Primary Containment~~ e

~~4.7.A.2. (Cont'd)~~ e

(A1)

~~d. Deleted~~ e

+

~~e. Deleted~~ e

+

~~f. Deleted~~ e

+



~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

SR 3.3.6.1.1

8. Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

Note: An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq (0.05 L_a)$ when tested at \geq Pa. (2) For door seal leakage, the overall air lock leakage rate is $\leq (0.02 L_a)$ when the air lock is pressurized to $(\geq 2.5$ psig for at least 15 minutes).

See Justification for Changes for BFN ISTS 3.6.1.2

3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(A6)

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

ACTION A

(M2) 1

(M2)
in MODE 3 in 12 hours
+ MODE 4 in 36 hours

ACTION B

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

See Justification for Changes for BFN 15TS 3.6.1.3



(A1)

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

~~j. Continuous Leak Rate Monitoring~~

(LC1)

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

k. The interior surfaces of the drywell and torus above the level one foot below the normal water line and outside surfaces of the torus below the water line shall be visually inspected each operating cycle for deterioration and any signs of structural damage with particular attention to piping connections and supports and for signs of distress or displacement.

(A3)

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

~~4.7.A Primary Containment~~

3.7.A.4 (Cont'd)

4.7.A.4 (Cont'd)

c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.

c. Once each operating cycle, each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.MM.

d. If Specifications 3.7.A.4.a, 3.7.A.4.b, or 3.7.A.4.c, cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.

SR 3.6.1.1.2

d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

See Justification for changes for BFN ISTS 3.6.1.7

(A1)

18 mos.

(AS)

5. Oxygen Concentration

5. Oxygen Concentration

a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100/psig, except as specified in 3.7.A.5.b.

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a Cold Shutdown condition within 24 hours.

c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

d. If the specifications of 3.7.A.5.a through 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

See Justification for changes for BFN ISTS 3.6.3.2



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.1 - PRIMARY CONTAINMENT

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The definition of PRIMARY CONTAINMENT INTEGRITY has been deleted from the proposed Technical Specifications. In its place the requirement for primary containment is that it "shall be OPERABLE." This was done because of the confusion associated with these definitions compared to its use in the respective LCO. The change is editorial in that all the requirements are specifically addressed in the proposed LCO for the primary containment along with the remainder of the LCOs in the Containment Systems Primary Containment subsection (e.g., air locks, isolation valves, suppression pool). Therefore, the change is purely a presentation preference adopted by the BWR Standard Technical Specifications, NUREG 1433.

- A3 CTS 4.7.A.2.k requirements for visual inspection of the drywell and torus surfaces are also contained in 10 CFR 50, Appendix J. These regulations require licensee compliance and cannot be revised by the licensee. These details of the regulations within CTS are repetitious and unnecessary. Therefore, the details also found in Appendix J have been deleted. This is considered a presentation preference and as such is considered an administrative change.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.1 - PRIMARY CONTAINMENT

- A4 CTS 3.7.A.2.b provides acceptance criteria for integrated leak rate testing, which is redundant to those contained in Primary Containment Leakage Rate Testing Program (CTS 6.8.4.3) requirements. The definition of L_a is provided in proposed BFN ISTS 1.1 and need not be repeated here. As such, this deletion is considered administrative.
- A5 The acceptance criteria for the leak test of the drywell to suppression chamber structure has been changed from 0.09 lb/sec of primary containment atmosphere at 1 psid to 0.25 inches of water for 10 minutes. Since these values are equivalent this is considered an administrative change.
- A6 CTS 4.7.A.2.h(1) requires repairs to be initiated immediately when it is determined the criterion of 4.7.A.2.g is exceeded. CTS 4.7.A.2.g requires LLRTs to be performed in accordance with the Primary Containment Leakage Rate Testing Program (CTS 6.8.4.3). CTS 4.7.A.2.h(2) then allows 48 hours to demonstrate 4.7.A.2.g can be met following detection of excessive local leakage. Since repairs are typically initiated immediately and proposed BFN ISTS ACTION A will only allow 1 hour to restore primary containment to OPERABLE status prior to requiring the initiation of a shutdown (reference Justification M2 below), CTS 4.7.A.2.h(1) has been deleted.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 CTS 3.7.A.2.a requires the primary containment to be OPERABLE at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the vessel. The proposed BFN ISTS 3.6.1.1 applicability is MODES 1, 2, and 3. This is more restrictive since CTS does not require the primary containment to be OPERABLE when in MODE 2, not critical and < 212°F.
- M2 Proposed Action A is more restrictive than CTS 3.7.A.2.c since the time allowed to reduce excessive nitrogen leakage prior to initiating a shutdown has been reduced from 8 hours to 1 hour. The time allotted to place the unit in Hot Shutdown (MODE 3) has been reduced from 16 hours to 12 hours. Proposed Action B requires the unit to be placed in Cold Shutdown (MODE 4), whereas, CTS 3.7.A.2.c only requires the unit to be placed in Hot Shutdown.

In addition, CTS 4.7.A.2.h.(2) allows 48 hours to demonstrate conformance to Appendix J following detection of excessive local leakage



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.1 - PRIMARY CONTAINMENT

and then requires a plant shutdown if conformance can not be demonstrated. CTS does not specify a completion time for shutdown and does not specify whether shutdown is to the Hot or Cold Shutdown Condition. The Proposed Actions A and B are more restrictive since they only allow 1 hour to restore primary containment and then require the unit be in MODE 3 in 12 and MODE 4 in 36 hours.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 The details relating to routine monitoring of plant status and operations parameters that reflect primary containment operability and the methods of performing this monitoring have been relocated to the Bases and procedures. Acceptance criteria for primary containment N₂ leakage (i.e., makeup consumption) have been relocated to procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.
- LC1 The continuous leak rate monitor does not necessarily relate directly to primary containment operability. In general, the BWR Standard Technical Specifications, NUREG 1433, do not specify indication-only or alarm-only equipment to be OPERABLE to support operability of a system or component. Control of the availability of, and necessary compensatory activities if not available for, indications, monitoring instruments, and alarms are addressed by plant operational procedures and policies. Therefore, the continuous leak rate monitor, and associated alarm surveillances and actions will be relocated to a licensee controlled document. Any changes will require a 10 CFR 50.59 evaluation.

PAGE 3 OF 3



UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A. Primary Containment~~

~~4.7.A. Primary Containment~~

(A1)

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN ISTS 3.6.1.1

M4 - LCO 3.6.1.2 Applicability

M1 - Proposed ACTIONS A+B

M1 - Proposed Note 1+2 to ACTIONS

M3 - Proposed SR 3.6.1.1.2

FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

SR 3.6.1.2.1

8. Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN 1STS 3.6.1.1

Note: An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

SR 3.6.1.2.1
Note

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq (0.05 L_a)$ when tested at \geq Pa. (2) For door seal leakage, the overall air lock leakage rate is $\leq (0.02 L_a)$ when the air lock is pressurized to (≥ 2.5 psig for at least 15 minutes).

See Justification for Changes for BFN 1STS 5.5.12.

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(A2)

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

Required Action C.2

Proposed Required Action C.1

M2

in MODE 3 in 12 hours
& MODE 4 in 36 hours

ACTION D

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

See Justification for Changes for BFN ISTS 3.6.1.3

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



FEB 22 1996

~~3.7/A.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A. Primary Containment~~

~~4.7.A. Primary Containment~~

(A1) 2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN ISTS 3.6.1.1

M4 - LCO 3.6.1.2 Applicability

M1 - Proposed ACTIONS A & B

Proposed Note 1+2 to ACTIONS

M3 - Proposed SR 3.6.1.1.2

FEB 22 1996

(A1)

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

SR 3.6.1.2.1

- 8. Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN ISTS 3.6.1.1

Note: An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

SR 3.6.1.2.1
Note

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq (0.05 L_a)$ when tested at \geq Pa. (2) For door seal leakage, the overall air lock leakage rate is $\leq (0.02 L_a)$ when the air lock is pressurized to $(\geq 2.5$ psig for at least 15 minutes).

See Justification for Changes for BFN ISTS 5.5.12

3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

A1

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

A2

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 15 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

Required Action C.2

Proposed Required Action C.1

ACTION D

M2

in MODE 3 in 12 hours & MODE 4 in 36 hours

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

See Justification for Changes for BFN ISTS 3.6.1.3



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A. Primary Containment~~

~~3.7.A. Primary Containment~~

(A1) 2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.c shall be taken.

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN ISTS 3.6.1.1

(M4) LCD 3.6.1.2 Applicability

Proposed ACTIONS A & B

(M1) Proposed Note 1 & 2 to Actions

(M3) Proposed SR 3.6.1.1.2



(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

~~SR 3.6.1.2.1~~

8. Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Change for BFN ISTS 3.6.1.1

Note: An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

SR 3.6.1.2.1 Note

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq (0.05 L_a)$ when tested at \geq Pa. (2) For door seal leakage, the overall air lock leakage rate is $\leq (0.02 L_a)$ when the air lock is pressurized to (\geq 2.5 psig for at least 15 minutes).

See Justification for Change for BFN ISTS 5.5.12

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

A2

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

Required Action C.2

M2

Proposed Required Action C.1

ACTION D

in Mode 3 in 12 hrs or mode 4 in 36 hrs

M2

See Justification For Changes for BFN ISTS 3.6.1.3 in this section

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.2 - PRIMARY CONTAINMENT AIR LOCK

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 CTS 4.7.A.2.h(1) requires repairs to be initiated immediately when it is determined the criterion of 4.7.A.2.g is exceeded. CTS 4.7.A.2.g requires LLRTs to be performed in accordance with the Primary Containment Leakage Rate Testing Program (CTS 6.8.4.3). CTS 4.7.A.2.h(2) then allows 48 hours to demonstrate 4.7.A.2.g can be met following detection of excessive local leakage. Since repairs are typically initiated immediately and proposed Required Action C.1 for 3.6.1.2 requires action be initiated to evaluate the primary containment overall leakage rate using the current air lock results and ACTION A of ISTS 3.6.1.1 will only allow 1 hour to restore primary containment to OPERABLE status prior to requiring the initiation of a shutdown (reference Justification M2 for Specification 3.6.1.1), CTS 4.7.A.2.h(1) has been deleted.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 The current requirements for the air lock are located within the primary containment TS requirements. The current definition of primary containment integrity requires only one air lock door to be closed and sealed (i.e., the seal mechanism intact and sealing the door). Thus, no actions are required if one door is inoperable provided the other door is OPERABLE, since primary containment integrity only requires the one door. The proposed LCO requires the entire air lock to be OPERABLE,



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.2 - PRIMARY CONTAINMENT AIR LOCK

which includes both doors, as well as the interlock mechanism and the leak-tightness of the barrel. ACTIONS are provided (proposed ACTIONS A and B) to ensure that if one door or its interlock mechanism is inoperable, the other door is closed, locked and periodically verified to be closed and locked. If the interlock mechanism is inoperable, an allowance is provided to open the door provided a dedicated individual controls the access. Notes are provided to allow the locked closed verification to be performed administratively if the door is in a limited access area. These two new actions are not applicable, however, if the entire air lock is inoperable (as stated in proposed Note 1 to both ACTIONS A and B). To ensure that the primary containment LCO will be entered if air lock leakage results in exceeding overall primary containment leakage, NOTE 2 to the ACTIONS is also included. Overall, these new ACTIONS provide additional restrictions to plant operation.

- M2 CTS 4.7.A.2.h requires repairs to be initiated immediately when it is determined that the criterion of 4.7.A.2.g is exceeded and if conformance to these criterion is not demonstrated within 48 hours following detection of excessive local leakage, a reactor shutdown is required. ACTION C of the proposed Specification requires the licensee to initiate action to evaluate primary containment overall leakage rate using the current air lock test results immediately, verify an air lock door closed within 1 hour and restore the air lock to OPERABLE status within 24 hours. If required ACTION C and the associate Completion Time is not met, the unit must be in MODE 3 in 12 hours and MODE 4 in 36 hours. This is more restrictive than current requirements.
- M3 This change adds a Surveillance to verify the interlock mechanism works properly (only one door can be opened at a time). This will ensure that one door is always closed which maintains containment integrity. The addition of new requirements represents a more restrictive change.
- M4 The current requirements for the air lock are located within the primary containment TS requirements (CTS 3.7.A.2.a), which requires the primary containment to be OPERABLE at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the vessel. The proposed BFN ISTS 3.6.1.2 applicability is MODES 1, 2, and 3. This is more restrictive since CTS does not require the primary containment to be OPERABLE when in MODE 2, not critical, and < 212°F.

UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~~~SURVEILLANCE REQUIREMENTS~~~~3.7.A. Primary Containment~~

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

LCO 3.6.1.3
Applicability

M2

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for changes
for BFN 15-TS 3.6.1.1



~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

See Justification for Changes for BFN ISTS 3.6.1.1+3.6.1.2

SR 3.6.1.3.10

1. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling

(A2) outage If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.



NOV 16 1992

(A1)

3.7.C. Secondary Containment

- 3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
- 4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

See Justification for Changes for BFN 1STs 3.6.4.1

(A1)

except Reactor Building Vacuum breakers

~~D. Primary Containment Isolation Valves~~

(M2)

Applicability

- 1. When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE except as specified in 3.7.D.2.

LCD 3.6.1.3

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

(L2)

Note 1 to ACTIONS
Note 2 to SR 3.6.1.3.2

~~D. Primary Containment Isolation Valves~~

(A1)

- 1. The primary containment isolation valves surveillance shall be performed as follows:
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation

(A2)

SR 3.6.1.3.5
SR 3.6.1.3.6
SR 3.6.1.3.7

Actual or

(L1)



NOV 16 1992

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

~~3.7.D. Primary Containment Isolation Valves~~

~~4.7.D. Primary Containment Isolation Valves~~

~~4.7.D.1.a (Cont'd)~~

(A1)

SR 3.6.1.3.5 and in accordance with Specification 1.0.MM, tested for closure times.

b. In accordance with Specification 1.0.MM, all normally open power operated primary containment isolation valves shall be functionally tested.

c. (Deleted)

(A2)

SR 3.6.1.3.8

d. At least once per operating cycle the OPERABILITY of the reactor coolant system instrument line flow check valves shall be verified.

EFCV actuates to the isolation position on a simulated instrument line break signal

(M4)

(L5) Proposed ACTION B

(L9) Proposed ACTION D

(M2) Proposed ACTION F

(A3) Proposed Note 2 to ACTIONS

(A4) Proposed Notes 3+4 to ACTIONS

2. In the event any primary containment isolation valve becomes inoperable, reactor operation may continue provided at least one valve, in each line having an inoperable valve, is OPERABLE and within 4 hours either:

(A5)

Condition A+C

- a. The inoperable valve is restored to OPERABLE status, or
- b. Each affected line is isolated by use of at least one deactivated containment isolation valve secured in the isolated position.

Required ACTION A1+C1

(L4)

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

ACTION E

(L8)

BFN Unit 1

The HOT SHUTDOWN CONDITION in 12 hours and in

(M1)

3.7/4.7-18

a closed manual valve, blind flange, or check valve with flow through the valve secured

(L3)

Proposed SRs 3.6.1.3.1
3.6.1.3.2
3.6.1.3.3
3.6.1.3.4
3.6.1.3.9

(M3)

2. Whenever a primary containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

Required ACTION A2+C2

(L6)

AMENDMENT NO. 189



(A1)

FEB 13 1995

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.7.F. Primary Containment Purge System

1. The primary containment purge system shall be OPERABLE for PURGING, except as specified in 3.7.F.2.
 - a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.
 - b. The results of laboratory carbon sample analysis shall show $\geq 85\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.
 - c. System flow rate shall be shown to be within $\pm 10\%$ of design flow when tested in accordance with ANSI N510-1975.
2. If the provisions of 3.7.F.1.a, b, and c cannot be met, the system shall be declared inoperable. The provisions of Technical Specification 1.C.1 do not apply. PURGING may continue using the Standby Gas Treatment System.

3. a. The 18-inch primary containment isolation valves associated with PURGING may be open during the RUN mode for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. The OPERABILITY of

SR 3.6.1.3.1
Note

(L7)

LCo 3.6.1.3

4.7.F. Primary Containment Purge System

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8.5 inches of water at system design flow rate ($\pm 10\%$).
 - a. The tests and sample analysis of Specification 3.7.F.1 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first or after 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.
 - b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
 - c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

see Justification for Changes for LTS 3.7.F/4.7.F in this Section

AMENDMENT NO. 215



APR 29 1991

3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

(A)

SURVEILLANCE REQUIREMENTS

3.7.F. Primary Containment Purge System

4.7.F. Primary Containment Purge System

3.7.F.3 (Continued)

LCD 3.6.1.3 these primary containment isolation valves is governed by Technical Specification 3.7.D.

b. Pressure control of the containment is normally performed by VENTING through 2-inch primary containment isolation valves which route effluent to the Standby Gas Treatment System. The OPERABILITY of these primary containment isolation valves is governed by Technical Specification 3.7.D.

(LAI)

3.7.G. Containment Atmosphere Dilution System (CAD)

4.7.G. Containment Atmosphere Dilution System (CAD)

1. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE with:

1. System Operability

a. Two independent systems capable of supplying nitrogen to the drywell and torus.

a. Cycle each solenoid operated air/nitrogen valve through at least one complete cycle of full travel in accordance with Specification 1.0.MM, and at least once per month verify that each manual valve in the flow path is open.

b. A minimum supply of 2,500 gallons of liquid nitrogen per system.

b. Verify that the CAD System contains a minimum supply of 2,500 gallons of liquid nitrogen twice per week.

See Justification for Change to BFN 1STS 3.6.3.1

3.7/4.7-22

AMENDMENT NO. 184

INSERT PROPOSED NEW SPECIFICATION 3.6.1.4

Insert new Specification 3.6.1.4, "Drywell Air Temperature," as shown in the BFN Unit 2 Improved Standard Technical Specifications.



JUSTIFICATION FOR CHANGES
BFN ISTS: 3.6.1.4 - DRYWELL AIR TEMPERATURE

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 A new Specification is being added requiring drywell air temperature to be $\leq 150^{\circ}\text{F}$. This is required since some accident analyses assume this temperature at the start of an accident. Appropriate ACTIONS and Surveillance Requirements are also added. This is consistent with the BWR Standard Technical Specifications, NUREG 1433.



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~~~LIMITING CONDITIONS FOR OPERATION~~

A1

~~SURVEILLANCE REQUIREMENTS~~~~3.7.A. Primary Containment~~

- 2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or, when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while

LCD 3.6.1.3
Applicability

M2

performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

- b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .
- c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes
for BFN 1STS 3.6.1.1

(A1)

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

- h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.
- (2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

See Justification for Changes for BFN ISTS 3.6.1.1 + 3.6.1.2

SR 3.6.1.3.10

- i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

(A2)



NOV 16 1992

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.7.C. Secondary Containment

- 3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
- 4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.4.1

(A1)

D. Primary Containment Isolation Valves

- 1. (When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE except as specified in 3.7.D.2.

(M2) Applicability

LCO 3.6.1.3

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

(L2)

Note 1 to ACTIONS
Note 2 to SR 3.6.1.3.2

BFN Unit 2

3.7/4.7-17

(A1)

except Reactor Building vacuum breakers

(A1)

D. Primary Containment Isolation Valves

- 1. The primary containment isolation valves surveillance shall be performed as follows:
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation

(A2)

SR 3.6.1.3.5
SR 3.6.1.3.6
SR 3.6.1.3.7

actual or (L1)

AMENDMENT NO. 204



~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.D. Primary Containment Isolation Valves~~

~~4.7.D. Primary Containment Isolation Valves~~

~~4.7.D.1.a (Cont'd)~~ (A1)

SR 3.6.1.3.5
SR 3.6.1.3.6

and in accordance with Specification 1.0.MM, tested for closure times.

b. In accordance with Specification 1.0.MM, all normally open power operated primary containment isolation valves shall be functionally tested.

c. (Deleted)

SR 3.6.1.3.8

d. At least once per operating cycle the OPERABILITY of the reactor coolant system instrument line flow check valves shall be verified. (A2)

(M4) EFCV actuates to the isolation position on a simulated instrument line break signal

(L5) Proposed ACTION B

(L9) Proposed ACTION D

(M2) Proposed ACTION F

(A3) Proposed Note 2 to ACTIONS

(A4) Proposed Notes 3+4 to ACTIONS

(A5) 2. In the event any primary containment isolation valve becomes inoperable, reactor operation may continue provided at least one valve, in each line having an inoperable valve, is OPERABLE and within 4 hours either:

Condition A+C

a. The inoperable valve is restored to OPERABLE status, or

b. Each affected line is isolated by use of at least one deactivated containment isolation valve secured in the isolated position. Required Action A.1+C.1

ACTIONS A+C

(L4)

Required Action A2+C.2

2. Whenever a primary containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily. (L6)

Proposed SRs 3.6.1.3.1, 3.6.1.3.2, 3.6.1.3.3, 3.6.1.3.4, 3.6.1.3.9 (M3)

(L3) a closed manual valve, blind flange, or check valve with flow through the valve secured

(ACTION E) 3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

BFN Unit 2

(M1) the HOT SHUTDOWN CONDITION in 12 hours and in

(L8)

FEB 13 1995

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.7.F. Primary Containment Purge System

- 1. The primary containment purge system shall be OPERABLE for PURGING, except as specified in 3.7.F.2.
 - a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.
 - b. The results of laboratory carbon sample analysis shall show $\geq 85\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.
 - c. System flow rate shall be shown to be within $\pm 10\%$ of design flow when tested in accordance with ANSI N510-1975.
- 2. If the provisions of 3.7.F.1.a, b, and c cannot be met, the system shall be declared inoperable. The provisions of Technical Specification 1.C.1 do not apply. PURGING may continue using the Standby Gas Treatment System.

T

SR3.6.1.3.1 Note

(L7)

3. a. The 18-inch primary containment isolation valves associated with PURGING may be open during the RUN mode for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. The OPERABILITY of

LCO 3.6.1.3

3.7/4.7-21

4.7.F. Primary Containment Purge System

- 1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8.5 inches of water at system design flow rate ($\pm 10\%$).
 - a. The tests and sample analysis of Specification 3.7.F.1 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first or after 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.
 - b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
 - c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

SEE JUSTIFICATION FOR CHANGES FOR CTS 3.7.F/4.7.F in 17; Section

AMENDMENT NO. 231

6 7



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.F. Primary Containment Purge System~~

(AI)

~~4.7.F. Primary Containment Purge System~~

~~3.7.F.3 (Continued)~~

LCo 3.6.1.3 these primary containment isolation valves is governed by Technical Specification 3.7.D.

b. Pressure control of the containment is normally performed by VENTING through 2-inch primary containment isolation valves which route effluent to the Standby Gas Treatment System. The OPERABILITY of these primary containment isolation valves is governed by Technical Specification 3.7.D.

(LAI)

~~3.7.G. Containment Atmosphere Dilution System (CAD)~~

1. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE with:

a. Two independent systems capable of supplying nitrogen to the drywell and torus.

b. A minimum supply of 2,500 gallons of liquid nitrogen per system.

~~4.7.G. Containment Atmosphere Dilution System (CAD)~~

1. System Operability

a. Cycle each solenoid operated air/nitrogen valve through at least one complete cycle of full travel in accordance with Specification 1.0.MM, and at least once per month verify that each manual valve in the flow path is open.

b. Verify that the CAD System contains a minimum supply of 2,500 gallons of liquid nitrogen twice per week.

BFN Unit 2

SEE JUSTIFICATION FOR CHANGE FOR BFN ISTS 3.6.3, 3.7/4.7-22

AMENDMENT NO. 197



INSERT PROPOSED NEW SPECIFICATION 3.6.1.4

Insert new Specification 3.6.1.4, "Drywell Air Temperature," as shown in the BFN Unit 2 Improved Standard Technical Specifications.

JUSTIFICATION FOR CHANGES
BFN ISTS: 3.6.1.4 - DRYWELL AIR TEMPERATURE

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 A new Specification is being added requiring drywell air temperature to be $\leq 150^{\circ}\text{F}$. This is required since some accident analyses assume this temperature at the start of an accident. Appropriate ACTIONS and Surveillance Requirements are also added. This is consistent with the BWR Standard Technical Specifications, NUREG 1433.



UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

(A1)

~~3.7.A. Primary Containment~~

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

LCD 3.6.1.3
Applicability

(112)

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.c shall be taken.

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes
for BFN 1STS 3.6.1.1



(A1)

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

See Justification for Changes for BFN ISTS 3.6.1.1 & 3.6.1.2

SR 3.6.1.3.10

1. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

(A2)



(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.7.C. Secondary Containment

- 3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
- 4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

See Justification for Changes for BFN ISTS 3.6.4.1

(A1)

~~D. Primary Containment Isolation Valves~~

m2
Applicability

- 1. When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE except as specified in 3.7.D.2.

LCO 3.6.1.3

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

(L2)

Note 1 to ACTIONS
Note 2 to SR 3.6.1.3.2

(A1)

except Reactor Building Vacuum breakers

(A1)

~~D. Primary Containment Isolation Valves~~

- 1. The primary containment isolation valves surveillance shall be performed as follows: (A2)
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation

SR 3.6.1.3.5
SR 3.6.1.3.6
SR 3.6.1.3.7

actual OR

(L1)



~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.D. Primary Containment Isolation Valves~~

~~4.7.D. Primary Containment Isolation Valves~~

(A1)

~~4.7.D.1.a (Cont'd)~~

(L5) Proposed Action B

(L9) Proposed Action D

(M2) Proposed Action F

(A3) Proposed Note 2 to Actions

(A4) Proposed Notes 3+4 to Actions

SR 3.6.1.3.5 and in accordance with Specification 1.0.MM, tested for closure times.

b. In accordance with Specification 1.0.MM, all normally open power operated primary containment isolation valves shall be functionally tested

c. (Deleted)

SR 3.6.1.3.8

d. At least once per operating cycle the OPERABILITY of the reactor coolant system instrument line flow check valves shall be verified.

(A2)

(M4) EPCV actuates to the isolation position on a Simulated Instrument line Break signal

(A5) 2. In the event any primary containment isolation valve becomes inoperable, reactor operation may continue provided at least one valve, in each line having an inoperable valve, is OPERABLE and within 4 hours either:

Condition A+C

a. The inoperable valve is restored to OPERABLE status, or

b. Each affected line is isolated by use of at least one deactivated containment isolation valve secured in the isolated position.

Required Action A.1 + C.1

ACTIONS A+C

2. Whenever a primary containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

Required Action A.2+C.2

Proposed SRs 3.6.1.3.1
3.6.1.3.2
3.6.1.3.3
3.6.1.3.4
3.6.1.3.9

(L3) a closed manual valve, blind Flange, or check valve with flow through the valve secured

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

Action E

(L6) (L8)

BFN Unit 3

the hot shutdown condition in 12 hours and in

3.7/4.7-18

(M1)

3.7.F. Primary Containment Purge System

4.7.F. Primary Containment Purge System

- 1. The primary containment purge system shall be OPERABLE for PURGING, except as specified in 3.7.F.2.
 - a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.
 - b. The results of laboratory carbon sample analysis shall show $\geq 85\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.
 - c. System flow rate shall be shown to be within $\pm 10\%$ of design flow when tested in accordance with ANSI N510-1975.
- 2. If the provisions of 3.7.F.1.a, b, and c cannot be met, the system shall be declared inoperable. The provisions of Technical Specification 1.C.1 do not apply. PURGING may continue using the Standby Gas Treatment System.

- 1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8.5 inches of water at system design flow rate ($\pm 10\%$).
 - a. The tests and sample analysis of Specification 3.7.F.1 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first or after 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.
 - b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
 - c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

- 3. a. The 18-inch primary containment isolation valves associated with PURGING may be open during the RUN mode for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. The OPERABILITY of

SR 3.6.1.3.1 Note

(L7)

See Justification for Changes for CTS 3.7.F/4.7.F in this Section



~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.F. Primary Containment Purge System~~

~~4.7.F. Primary Containment Purge System~~

~~3.7.F.3 (Continued)~~

LCO 3.6.1.3 these primary containment isolation valves is governed by Technical Specification 3.7.D.

b. Pressure control of the containment is normally performed by VENTING through 2-inch primary containment isolation valves which route effluent to the Standby Gas Treatment System. The OPERABILITY of these primary containment isolation valves is governed by Technical Specification 3.7.D.

LA1

3.7.G. Containment Atmosphere Dilution System (CAD)

4.7.G. Containment Atmosphere Dilution System (CAD)

1. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE with:

1. System Operability

a. Two independent systems capable of supplying nitrogen to the drywell and torus.

a. Cycle each solenoid operated air/nitrogen valve through at least one complete cycle of full travel in accordance with Specification 1.0.MM, and at least once per month verify that each manual valve in the flow path is open.

b. A minimum supply of 2,500 gallons of liquid nitrogen per system.

b. Verify that the CAD System contains a minimum supply of 2,500 gallons of liquid nitrogen twice per week.

See Justification for Changes for BFN 15TS 3.6.3.1

INSERT PROPOSED NEW SPECIFICATION 3.6.1.4

Insert new Specification 3.6.1.4, "Drywell Air Temperature," as shown in the BFN Unit 2 Improved Standard Technical Specifications.



JUSTIFICATION FOR CHANGES
BFN ISTS: 3.6.1.4 - DRYWELL AIR TEMPERATURE

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 A new Specification is being added requiring drywell air temperature to be $\leq 150^{\circ}\text{F}$. This is required since some accident analyses assume this temperature at the start of an accident. Appropriate ACTIONS and Surveillance Requirements are also added. This is consistent with the BWR Standard Technical Specifications, NUREG 1433.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

In addition, the PCIV LCO now exempts the reactor building-to-suppression chamber vacuum breakers and scram discharge volume vent and drain valves since they are governed by other LCOs. Any changes to the requirements for these valves are discussed in the new LCO Justification for Changes.

- A2 The current technical specification (CTS) 4.7.D.1.a frequency of "once per operating cycle" has been changed to "In accordance with the Inservice Testing Program" for proposed SR 3.6.1.3.5 (stroke time tests). The CTS 4.7.D.1.d frequency of "once per operating cycle" has been changed to "18 months" for proposed SR 3.6.1.3.8. Since an operating cycle is 18 months and the current IST program requires testing every 18 months, this change is considered administrative in nature. The CTS 4.7.A.2.i frequency of "each refueling outage" has been replaced with "in accordance with the Primary Containment Leakage Rate Testing Program" for SR 3.6.1.3.10. This program requires Appendix J requirements to be met. The Appendix J requirements will always supersede the Technical Specification requirements (unless an exemption is approved) since Appendix J is the rule. Therefore, this change is purely an administrative preference in presentation.
- A3 This proposed Note ("Separate Condition entry is allowed for each penetration flow path") provides explicit instructions for proper application of the actions for Technical Specification compliance. In



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

conjunction with the proposed Specification 1.3 - "Completion Times," this Note provides direction consistent with the intent of the existing Actions for inoperable isolation valves.

- A4 The proposed ACTIONS include Notes 3 and 4. These Notes facilitate the use and understanding of the intent to consider any system affected by inoperable isolation valves, which is to have its ACTIONS also apply if it is determined to be inoperable. Note 4 clarifies that these "systems" include the primary containment. With proposed LCO 3.0.6, this intent would not necessarily apply. This clarification is consistent with the intent and interpretation of the existing Technical Specifications, and is therefore considered an administrative presentation preference.
- A5 The current single Action for "any primary containment isolation valve" has been divided into three ACTIONS. Proposed ACTION A for one valve inoperable in a penetration that has two valves, proposed ACTION B for two valves inoperable in a penetration that has two valves, and proposed ACTION C for one valve inoperable in a penetration that has only one valve. All technical changes are discussed elsewhere in this section. As such, this change is considered an administrative presentation preference.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 CTS 3.7.D.3 requires an orderly shutdown be initiated and the reactor to be in the COLD SHUTDOWN CONDITION within 24 hours when certain conditions can not be met. Proposed Action E will require the plant be in MODE 3 in 12 hours and MODE 4 in 36 hours. The addition of this intermediate step to the COLD SHUTDOWN CONDITION is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.
- M2 CTS applicability for PCIV operability is when primary containment integrity is required. Per CTS 3.7.A.2.a, primary containment integrity is required at all times the reactor is critical or when the reactor water temperature is > 212°F and fuel is in the reactor vessel. The proposed applicability of MODES 1, 2, and 3 is more restrictive since CTS does not require primary containment integrity when in MODE 2, not

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

critical and < 212°F. The proposed Specification is also applicable when associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, which adds a MODE 4 and 5 requirement for the RHR Shutdown Cooling isolation valves. An appropriate ACTION has been added (proposed ACTION F) for when the valves cannot be isolated (since the unit is already in MODE 4 or 5, the current actions provide no appropriate compensatory measures). ACTION F requires the licensee to initiate action to suspend operations with the potential for draining the reactor vessel immediately and to restore valve(s) to OPERABLE status immediately. If suspending an OPDRV would result in closing the RHR Shutdown Cooling valves, an alternative required action is provided to immediately initiate action to restore the valves to OPERABLE status.

- M3 New Surveillance Requirements have been added. SRs 3.6.1.3.1, 3.6.1.3.2 and 3.6.1.3.3 ensure PCIVs are in their proper position or state. SRs 3.6.1.3.4 and 3.6.1.3.9 ensure the traversing incore probe (TIP) squib valves will actuate if required. These SRs are additional restrictions on plant operation.
- M4 This change adds acceptance criteria to the Surveillance Requirement which requires an Operability test of the instrument line excess flow check valves (EFCVs). The acceptance criteria added requires that the EFCVs actuate to the isolation position on a simulated instrument line break signal. The addition of acceptance criteria which did not previously exist in Technical Specifications constitutes a more restrictive change.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 CTS 3.7.F.3.b provides no requirements, it just explains that the normal method of containment pressure control is through 2-inch PCIVs, which route effluent through the SGTS. Since the OPERABILITY of these valves is governed by proposed BFN ISTS 3.6.1.3, the specification provides no requirements and has been eliminated. Any details relating to PCIV operability have been relocated to the Bases of LCO 3.6.1.3. Placing these details in the Bases provides assurance they will be appropriately maintained since changes to these details will require a 50.59 evaluation.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

"Specific"

- L1 The phrase "actual or" in reference to the automatic isolation signal, has been added to the Surveillance Requirement for verifying that each PCIV actuates on an automatic isolation signal. This allows satisfactory automatic PCIV isolations for other than Surveillance purposes to be used to fulfill the Surveillance Requirements. Operability is adequately demonstrated in either case since the PCIV cannot discriminate between "actual" or "simulated".
- L2 The provisions of the "*" Note of CTS 3.7.D.1 are encompassed by Note 1 to the ACTIONS, which allows penetration flow paths to be unisolated intermittently under administrative controls (except for the 18-inch purge valve penetration flow paths). However, the ISTS allowance applies to all primary containment isolation valves (except for 18-inch purge valve penetration flow paths) not just locked or sealed closed valves. The allowance is presented in proposed ACTIONS Note 1 and in SR 3.6.1.3.2, Note 2. Opening of primary containment penetrations on an intermittent basis is required for performing surveillances, repairs, routine evolutions, etc.
- L3 CTS 3.7.D.2.b allows isolating the primary containment penetrations with at least one deactivated valve secured in the isolated position when one PCIV is inoperable. The proposed ACTIONS A and C of LCO 3.6.1.3 allow the use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve (for Condition A only) with flow through the valve secured. The Action utilizing a deactivated automatic or manual valve is appropriate on the basis that these isolations present a boundary which is not affected by a single failure. The ability to utilize the valves downstream of the outboard PCIVs is an acceptable isolation since it meets the acceptance criteria of not being affected by a single active failure.
- L4 CTS 3.7.D.2 allows reactor operation to continue when any PCIV becomes inoperable provided that at least one valve in each line having an inoperable valve is operable and within 4 hours the affected line is isolated or the inoperable valve is restored to OPERABLE status. Based on the wording, this only applies to lines with two isolation valves. This is equivalent to proposed ACTION A, however, the proposed ACTION allows additional time to isolate the main steam lines. A Completion Time of 8 hours for the MSLs allows a period of time to restore the MSIVs to OPERABLE status given the fact that MSIV closure will result in isolation of the MSLs and a potential for plant shutdown. For



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

penetration flow paths with only one PCIV, proposed ACTION C allows 4 hours to restore an inoperable valve to OPERABLE status and 12 hours to restore EFCVs in reactor instrumentation line penetrations. The four hour Completion Time is reasonable considering the relative stability of the closed system to act as a penetration boundary and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3. The Completion Time of 12 hours is reasonable considering the instrument and the small pipe diameter of the affect penetrations. During the allowed time, a limiting event would still be assumed to be within the bounds of the safety analysis, assuming no single active failure. Allowing this extended time to potentially avoid a plant transient caused by the immediate forced shutdown is reasonable based on the probability of an event and does not represent a significant decrease in safety.

- L5 In the event both valves in a penetration are inoperable, the existing Specification, which requires maintaining one isolation valve OPERABLE, would not be met and an immediate shutdown is required. The proposed ACTION (ACTION B) provides 1 hour prior to commencing a required shutdown. This proposed 1 hour period is consistent with the proposed BWR Standard Technical Specification time allowed for conditions when the primary containment is inoperable. The proposed change will provide consistency in actions for these various containment degradations.
- L6 The frequency of the periodic verification required when a penetration has been isolated to comply with current Specification 3.7.D.2 has been changed from daily to monthly. These valves are strictly controlled and are operated in accordance with plant procedures. Daily verification that these valves are still isolated places an undue burden on plant operations and provides little if any gain in safety, since these valves are rarely found in the unisolated condition, once closed. Note that CTS 4.7.D.2 requires the position of one other valve in the line be "recorded" daily versus the ISTS wording of "verified." ISTS also allows an inoperable valve to be used for isolating the penetration.
- L7 The Note to SR 3.6.1.3.1 allows the SR to not be met (i.e., do not have to verify closed) when the valves are open for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry and for Surveillances that require the valves to be open. For these reasons, it is deemed acceptable to open the valves for short periods of time. CTS 3.7.F.3.a, which allows the 18-inch primary containment isolation valves associated with PURGING to be open during the RUN mode during a 24-hour period after entering the RUN mode and/or

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

for a 24-hour period prior to entering the SHUTDOWN mode, is encompassed by the provisions of the Note. The additional exemptions allowed by the Note are acceptable since the 18-inch purge valves continue to be capable of closing in the environment following a LOCA.

- L8 The time allowed to shutdown the plant when the required actions are not met has been changed from "in the COLD SHUTDOWN CONDITION within 24 hours" to in MODE 3 (Hot Shutdown) in 12 hours and MODE 4 (Cold Shutdown) within 36 hours. The proposed allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The additional 12 hours allowed to reach Mode 4 is offset by the safety benefit of being subcritical (MODE 3) in a shorter required time.
- L9 This change adds proposed ACTION D which relaxes the allowed outage time from 4 hours to 8 hours to isolate the affected penetration if one main steam isolation valve (MSIV) in one or more penetrations is inoperable (due to leakage or other reason). This will allow a longer period of time to restore the MSIVs to OPERABLE status in order to prevent the potential for a plant shutdown by isolating the main steam line(s). During the additional time allowed, a limiting event would still be assumed to be within the bounds of the safety analysis, assuming no single active failure. Allowing this extended time to potentially avoid a plant transient caused by a plant shutdown is reasonable and does not represent a significant decrease in safety.



UNIT 1

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

2007 / 10 / 14



~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A PRIMARY CONTAINMENT~~

~~4.7.A PRIMARY CONTAINMENT~~

~~3. Pressure Suppression Chamber
Reactor Building Vacuum Breakers~~

~~3. Pressure Suppression Chamber
Reactor Building Vacuum Breakers~~

Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.

The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.

M1

Applicability Modes 2 + 3

SR 3.6.1.5.3

A2

Proposed Note for ACTIONS

ACTIONS A & C

L1

Proposed ACTIONS B, D & E

L1

From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.

SR 3.6.1.5.3

L2

SR 3.6.1.5.3

LAI

A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

M2

Proposed SR 3.6.1.5.1

~~4. Drywell-Pressure Suppression Chamber Vacuum Breakers~~

~~4. Drywell-Pressure Suppression Chamber Vacuum Breakers~~

a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c., below.
b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.
b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

BFN Unit 1

See Justification for Changes for BFN ISTS 3.6.1.6

3.7/4.7-10

AMENDMENT NO. 2 2 2



L2

TABLE 3.7.A
INSTRUMENTATION FOR CONTAINMENT SYSTEMS

Minimum No.
Operable Per
Trip System

2

Function

Instrument Channel -
Pressure suppression
chamber-reactor building
vacuum breakers
(PdIS-64-20, 21)

Trip Level Setting

0.5 psid

Action

(1)

Remarks

Actuates the pressure
suppression chamber-
reactor building
vacuum breakers.

Footnote:

(1) - Repair in 24 hours. If the function is not OPERABLE in 24 hours, declare the system or component inoperable.

BEN
Unit 1

3.7/4.7-24a

PAGE 3 OF 4

AMENDMENT NO. 222

Specification 3.6.1.5
JUL 17 1985

3



L2

TABLE 4.7.A

CONTAINMENT SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>Function</u>	<u>Functional Test</u>	<u>Calibration</u>	<u>Instrument Check</u>
Instrument Channel- Pressure suppression chamber reactor building vacuum breakers (Fdis-64-20, 21)	Once/month (1)	Once/18 months (2)	None

Footnotes:

- (1) - Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify OPERABILITY of the trip and alarm functions.
- (2) - Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the level setting.

REF
Unit 1

3.7/4.7-24b

AMENDMENT NO. 222
PAGE 4 OF 4

JUL 17 1995

Specification 3.6.1.5



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.A Primary Containment (AI)

4.7.A Primary Containment

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

(AI)
a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.

SR 3.6.1.5.
a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.

(MI)
Applicability Modes 1, 2 & 3

SR 3.6.1.5.3

(A2) Proposed Note for ACTIONS →

SR 3.6.1.5.3

(LI)

ACTIONS A+C

b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.

(LAI)
SR 3.6.1.5.3
b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

(LI)

Proposed ACTIONS B, D + E

(M2)

Proposed SR 3.6.1.5.1 →

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c., below.
b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.
b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

See Justification for Changes for BFN ISTS 3.6.1.6



42

TABLE 3.7.A

INSTRUMENTATION FOR CONTAINMENT SYSTEMS

Minimum No.
Operable Per
Trip System

2

Function

Instrument Channel -
Pressure suppression
chamber-reactor building
vacuum breakers
(PdIS-64-20, 21)

Trip Level Setting

0.5 psid

Action

(1)

Remarks

Actuates the pressure
suppression chamber-
reactor building
vacuum breakers.

Footnote:

(1) - Repair in 24 hours. If the function is not OPERABLE in 24 hours, declare the system or component inoperable.

BFN
Unit 2

3.7/4.7-24a

AMENDMENT NO. 237
PAGE 3 OF 4

Specification 3.6.1.5
JUL 17 1995



TABLE 4.7.A

CONTAINMENT SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>Function</u>	<u>Functional Test</u>	<u>Calibration</u>	<u>Instrument Check</u>
Instrument Channel- Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21)	Once/month ⁽¹⁾	Once/18 months ⁽²⁾	None

Footnotes:

- (1) - Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify OPERABILITY of the trip and alarm functions.
- (2) - Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the level setting.

L2

BRN
Date 2

3.7/4.7-24b

PAGE 4 OF 4

AMENDMENT NO. 237

Specification 3.6.15
JUL 17 1985

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~2.7.A PRIMARY CONTAINMENT~~

(A1)

~~3. Pressure Suppression Chamber
Reactor Building Vacuum Breakers~~

(A1)

a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.

(M1) Applicability notes 1, 2, 3

SR 3.6.1.5.3

(A2) Proposed Note for Actions

(L1) Actions A+C

(L1) Proposed Actions B, D, +E

b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c below.
b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

~~4.7.A PRIMARY CONTAINMENT~~

~~3. Pressure Suppression Chamber
Reactor Building Vacuum Breakers~~

SR 3.6.1.5

a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.

SR 3.6.1.5.3

(L2)

SR 3.6.1.5.3

(LAI)

b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

(M2)

Proposed SR 3.6.1.5.1

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.
b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

L2

TABLE 3.7.A

INSTRUMENTATION FOR CONTAINMENT SYSTEMS

Minimum No. Operable Per Trip System	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21)	0.5 psid	(1)	. Actuates the pressure suppression chamber-reactor building vacuum breakers.

Footnote:

(1) - Repair in 24 hours. If the function is not OPERABLE in 24 hours, declare the system or component inoperable.

BFN
Under 3

3.7/4.7-23b

AMENDMENT NO. 196

3 OF 4

Specification 3.6.1.5
JUL 17 1995

TABLE 4.7.A

CONTAINMENT SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>Function</u>	<u>Functional Test</u>	<u>Calibration</u>	<u>Instrument Check</u>
Instrument Channel- Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21)	Once/month ⁽¹⁾	Once/18 months ⁽²⁾	None

Footnotes:

- (1) - Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify OPERABILITY of the trip and alarm functions.
- (2) - Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the level setting.

(12)

Specification 3.6.1.5
JUL 17 1995

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.5
REACTOR BUILDING-TO-SUPPRESSION CHAMBER VACUUM BREAKERS

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 Existing LCO 3.7.A.3 is being replaced by proposed LCO 3.6.1.6. The proposed LCO will contain a Note stating that: "Separate Condition entry is allowed for each line." This note clarifies that the Conditions and Required Actions that follow may be applied to each of the two reactor building-to-suppression chamber vent paths without regard to vent path status. Each vent path contains two vacuum breakers in series. This note provides directions consistent with the intent of the Required Actions. This change is consistent with NUREG-1433.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 CTS 3.7.A.2.a requires the primary containment to be OPERABLE at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the vessel. The proposed BFN ISTS 3.6.1.6 applicability is MODES 1, 2, and 3. This is more restrictive since CTS does not require the primary containment to be OPERABLE when in MODE 2, not critical and < 212°F.
- M2 A new Surveillance Requirement has been added to verify each vacuum breaker is closed (except when they are open for performance of Surveillances) every 14 days. This is consistent with the BWR Standard Technical Specifications, NUREG 1433.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.5
REACTOR BUILDING-TO-SUPPRESSION CHAMBER VACUUM BREAKERS

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 Details of visual inspections of valves have been relocated to plant procedures. This type of inspection is more appropriately controlled by plant procedures. The valves are still required by Technical Specifications to be cycled and their setpoint verified to ensure operability. Any changes to procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 Existing LCO 3.7.A.3.b identifies the currently required actions if one reactor building-to-suppression chamber vacuum breaker is inoperable. If more than one vacuum breaker is inoperable, the existing specification assumes either containment integrity is lost or the ability to relieve negative pressure in the containment is lost. Therefore, LCO 3.7.A.3.b. defaults to 1.0.C.1 which requires that the reactor be placed in Hot Standby within 6 hours and Cold Shutdown within the following 30 hours. Proposed LCO 3.6.1.6 recognizes that there are two vacuum breakers in series in each of two vent paths between the reactor building and suppression chamber. As a result, if one vacuum breaker in each vent path is not closed (Condition A), containment integrity and venting capability are still maintained and 7 days is provided to restore the redundancy for containment integrity in each vent line. Likewise, if two vacuum breaker valves in one vent line are inoperable but closed (Condition C), containment integrity and venting capability are still maintained and 7 days is provided to restore the redundant vent path. Therefore, proposed Specification 3.6.1.6 makes the distinction between loss of redundancy and loss of function. The existing specification fails to make this distinction between loss of function and loss of redundancy and, therefore, is unnecessarily conservative. In addition, loss of function (loss of containment integrity (Condition B) or loss of venting capability (Condition D)) will require initiating action within 1 hour instead of immediately. Also, CTS 3.7.A.3 does not have a specific shutdown requirement, therefore, CTS 1.0.C.1 applies. CTS 1.0.C.1 requires the unit be placed in Hot Standby within 6 hours and Cold Shutdown within the following 30 hours. Proposed ACTION E requires the Unit to be placed in Hot Shutdown with 12 hours and Cold Shutdown within 36 hours. Proposed ACTION E is considered less restrictive since additional time is allowed prior to requiring the plant to be in a lesser Mode (i.e., Proposed Action E requirement to be in Hot Shutdown in 12 hours versus the CTS requirement to be in Hot Standby in 6 hours). This change is consistent with NUREG-1433.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.5
REACTOR BUILDING-TO-SUPPRESSION CHAMBER VACUUM BREAKERS

- L2 The vacuum breaker actuation instrumentation Surveillances are proposed to be deleted from Technical Specifications. The requirement of SR 3.6.1.5.3 to ensure the vacuum breakers are full open at 0.5 psid is sufficient. Vacuum breaker actuation instrumentation is required to be OPERABLE to satisfy the setpoint verification Surveillance Requirement (SR 3.6.1.5.3) for the vacuum breakers. If the vacuum breaker actuation instrumentation is inoperable, then the Surveillance Requirement cannot be satisfied and the appropriate actions must be taken for inoperable vacuum breakers in accordance with the ACTIONS of Specification 3.6.1.5. As a result, the requirements for the vacuum breaker actuation instrumentation are adequately addressed by the requirements of Specification 3.6.1.5 and SR 3.6.1.5.3 and are proposed to be deleted from Technical Specifications.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



(A1)

3.7.A PRIMARY CONTAINMENT

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

- a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.
- b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.

4.7.A PRIMARY CONTAINMENT

3. Pressure Suppression Chamber- Reactor Building Vacuum Breakers

- a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.
- b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

See Justification for Changes for BFN ISTS 3.6.1.5

~~4. Drywell Pressure Suppression Chamber Vacuum Breakers~~

(A1)

(M1) Applicability Modes 1, 2 & 3

LCO 3.6.1.6

When performing their intended function

(A2) Note 2 to SR 3.6.1.6.1

- a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c, below. (A1)
- b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

~~4. Drywell Pressure Suppression Chamber Vacuum Breakers~~

SR 3.6.1.6.2

- a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.

(M2) Proposed SR 3.6.1.6.1 →

- b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

(L3)

BFN Unit 1

(L1) Proposed ACTION A → 3.7/4.7-10

(L2) Proposed ACTION B →

AMENDMENT NO. 222



(A1)

~~3.7.4.7-11~~
~~OPERATING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

~~4.7.A Primary Containment~~

~~3.7.A.4 (Cont'd)~~

~~4.7.A.4 (Cont'd)~~
SR 3.6.1.6.3

LC03.6.1.6 c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.

c. Each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.224.

See Justification for Changes for BFN 1ST3 B.6.1.1

ACTION C
d. If Specifications 3.7.A.4.a, 3.7.A.4.b, or 3.7.A.4.c cannot be met, the unit shall be placed in a COLD SHUTDOWN CONDITION in an orderly manner within 24 hours.
36-44
In a HOT SHUTDOWN Condition in 12hrs
9nd

d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

5. Oxygen Concentration

5. Oxygen Concentration

a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. Within the 24-hour period subsequent to placing the reactor in the RUN MODE following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a COLD SHUTDOWN CONDITION within 24 hours.

c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

d. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN CONDITION within 24 hours.

See Justification for CHANGES FOR BFN 1ST3 3.6.3.1



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



JUL 17 1995

3.7/4.7 CONTAINMENT SYSTEMS

(A1)

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.A Primary Containment

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

- a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.
- b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.

4.7.A Primary Containment

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

- a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.
- b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

See Justification for Changes for BFN 15TS 3.6.1.5

(A1) Drywell-Pressure Suppression Chamber Vacuum Breakers

(M1)

Applicability MODES 6, 2 + 3

LCO 3.6.1.6

When performing their intended function

(A2)

Note 2 to SR 3.6.1.6.1

a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c., below.

b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

(A1) Drywell-Pressure Suppression Chamber Vacuum Breakers

SR 3.6.1.6.2

(M2)

Proposed SR 3.6.1.6.1

a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.

(L3)

b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

BFN (L1) Proposed ACTION A →

Unit 2

(L2) Proposed ACTION B →

3.7/4.7-10

AMENDMENT NO. 237

PAGE 2 OF 3

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

~~4.7.A Primary Containment~~

~~3.7.A.4 (Cont'd)~~

~~4.7.A.4 (Cont'd)~~

SR 3.6.1.63

LC036.16 c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.

c. Each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches

in accordance with Specification 1.0.YM

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.1.1

ACTION 2

d. If Specifications 3.7.A.4.a, .b, or .c cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.

d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

L4 24 36

in a Hot Shutdown condition in 12 hrs

M3

5. Oxygen Concentration

5. Oxygen Concentration

a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100/psig, except as specified in 3.7.A.5:b.

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a Cold Shutdown condition within 24 hours.

c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

d. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.3.1

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

JUL 17 1985

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

3.7.A PRIMARY CONTAINMENT

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

- a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.
- b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.

(A1)

~~4. Drywell-Pressure Suppression Chamber Vacuum Breakers~~

(M1)

Applicability Modes 1,2+3

LCO 3.6.1.6

When performing their intended function

- a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c below.

(A1)

(A2)

Note 2 to SR 3.6.1.6.1

- b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

BFN Unit 3

(L1)

Proposed Action A

(L2)

Proposed Action B

3.7/4.7-10

4.7.A PRIMARY CONTAINMENT

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

- a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.
- b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

See Justification for Changes for BFN ISTS 3.6.1.5

(A1)

~~4. Drywell-Pressure Suppression Chamber Vacuum Breakers~~

SR 3.6.1.6.2

- a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.

(M2)

Proposed SR 3.6.1.6.1

- b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

(L3)

AMENDMENT NO. 196

PAGE 2 OF 3

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

~~4.7.A Primary Containment~~

~~3.7.A.4 (Cont'd)~~

~~4.7.A.4 (Cont'd)~~

SR 3.6.1.6.3

LCO 3.6.1.6 c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.

c. Once each operating cycle, each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches

(A3) in accordance with Specification 1.0.MM.

Action C d. If Specifications 3.7.A.4.a, 3.7.A.4.b, or 3.7.A.4.c, cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.

d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

(M3)
30-14
In a HOT SHUT DOWN condition in 12 hrs and

See Justification for Changes for BFN 15 TS 3.6.1.1

5. Oxygen Concentration

5. Oxygen Concentration

a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100/psig, except as specified in 3.7.A.5.b.

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a Cold Shutdown condition within 24 hours.

c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

d. If the specifications of 3.7.A.5.a through 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

See Justification For Changes 15 TS 3.6.3.1

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.6
SUPPRESSION-CHAMBER-TO-DRYWELL VACUUM BREAKERS

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The words "except when performing their intended function" have been added to preclude requiring the LCO to be met when the valves cycle automatically. Since their intent is to open when a sufficient differential pressure exists, this change is considered administrative only.
- A3 CTS 4.7.A.4.c is performed in accordance with the Inservice Testing Program on a frequency of every operating cycle. Proposed SR 3.6.1.6.3 is to be performed every 18 months. Since an operating cycle at BFN is approximately 18 months, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 CTS 3.7.A.2.a requires the primary containment to be OPERABLE at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the vessel. The proposed BFN ISTS 3.6.1.6 applicability is MODES 1, 2, and 3. This is more restrictive since CTS does not require the primary containment to be OPERABLE when in MODE 2, not critical and < 212°F.
- M2 A new Surveillance Requirement (proposed SR 3.6.1.6.1) has been added to verify the vacuum breakers are closed once every 14 days. This new SR ensures the "closed" requirement of the LCO statement is being met. This is an additional restriction on plant operation.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.6
SUPPRESSION-CHAMBER-TO-DRYWELL VACUUM BREAKERS

- M3 CTS 3.7.A.4.d requires an orderly shutdown be initiated and the reactor to be in the Cold Shutdown Condition within 24 hours when certain conditions can not be met. Proposed Action C will require the plant be in MODE 3 (Hot Shutdown Condition) in 12 hours and MODE 4 (Cold Shutdown Condition) in 36 hours. The addition of this intermediate step to the Cold Shutdown Condition is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Specific"

- L1 Proposed ACTION A allows 72 hours to restore an inoperable vacuum breaker to OPERABLE status, with one of the required vacuum breakers inoperable for opening. This is allowed since the remaining nine OPERABLE breakers are capable of providing the vacuum relief function. The 72 hours is considered acceptable due to the low probability of an event in conjunction with an additional failure in which the remaining vacuum breaker capability would not be adequate.
- L2 Proposed Action B allows a short time to close an open vacuum breaker since there is low probability of an event that would pressurize primary containment. An open vacuum breaker allows communication between the drywell and suppression chamber airspace and, as a result, there is the potential for suppression chamber overpressurization due to this bypass leakage if a LOCA were to occur. If vacuum breaker position indication is not reliable, an alternate method of verifying that the vacuum breakers are closed is to verify that a differential pressure of 0.5 psid between the suppression chamber and drywell is maintained for 1 hour without makeup. The required 2 hour Completion Time is considered adequate to perform this test.
- L3 Existing Specification 4.7.A.4.b requires that "When it is determined that a vacuum breaker is inoperable for opening at a time when operability is required, all other vacuum breakers shall be exercised immediately and every 15 days thereafter until the inoperable vacuum breaker has been returned to normal service." This requirement is not included in NUREG-1433 and will be deleted. This change eliminates the requirement to demonstrate the OPERABILITY of the redundant vacuum breakers whenever a vacuum breaker is declared inoperable. This change acknowledges that the inoperability of a vacuum breaker is not automatically indicative of a similar condition in the redundant vacuum breakers unless a generic failure is suspected and that the periodic

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.6
SUPPRESSION-CHAMBER-TO-DRYWELL VACUUM BREAKERS

frequencies specified to demonstrate OPERABILITY have been shown to be adequate to ensure equipment OPERABILITY. Therefore, this change allows credit to be taken for normal periodic surveillance as a demonstration of OPERABILITY and availability of the remaining components and reduces unnecessary challenges and wear to redundant components. This change is consistent with NUREG-1433.

- L4 CTS 3.7.A.4.d requires the unit to be placed in a Cold Shutdown condition in an orderly manner within 24 hours. Proposed ACTION C is less restrictive since it requires the unit to be placed in MODE 3 (Hot Shutdown condition) in 12 hours and in MODE 4 (Cold Shutdown condition) in 36 hours. The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP





AUG 23 1991

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

A1

3.7.A. Primary Containment

Proposed Required Action A1

M3

3.7.A.1 (Cont'd)

ACTION A

Condition A

c. With the suppression pool water temperature > 95°F initiate pool cooling and restore the temperature to ≤ 95°F within 24 hours or be in at least HOT SHUTDOWN CONDITION within the next 6 hours and in the COLD SHUTDOWN CONDITION within the following 30 hours.

LCD 3.6.2.1.a

Required Action A.2

L1

ACTION B

Required Action D.3

When any OPERABLE IRM Channel is ≤ 25/40 divisions of full scale on Range 7

ACTION C

Condition C

d. With the suppression pool water temperature > 105°F during testing of ECCS or relief valves, stop all testing, initiate pool cooling and follow the action in Specification 3.7.A.1.c above.

LCD 3.6.2.1.b

Required Action C.1

LA1

ACTION A

ACTION D

Condition D

e. With the suppression pool water temperature > 110°F during the STARTUP CONDITION, HOT STANDBY CONDITION (with all control rods not inserted), or REACTOR POWER OPERATION, the reactor shall be scrammed.

LCD 3.6.2.1.c

Required Action D.1

Applicability Modes 1, 2 + 3

M4

Proposed Required Action D.2

ACTION E

Condition E

f. With the suppression pool water temperature > 120°F following reactor isolation, depressurize to < 200 psig at normal cooldown rates.

M5

M4

UNIT 2

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



~~3.7/A.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

(A1)

~~Specification~~

~~A. Primary Containment~~

- 1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

Applicability

LC0 3.6.2.1

- a. Minimum water level = -6.25" (differential pressure control >0 psid)
-7.25" (0 psid differential pressure control)

- b. Maximum water level = -1"

(A1)

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

~~Specification~~

~~A. Primary Containment~~

~~1. Pressure Suppression Chamber~~

- a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

SEE JUSTIFICATION FOR CHANGES FOR BFN 1153.5.2

SEE JUSTIFICATION FOR CHANGES FOR BFN 3.6.2.2

(M2)

Add SR 3.6.2.1.1 first frequency - once / 24 hours



LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.A. Primary Containment

3.7.A.1 (Cont'd)

Proposed Required Action A 1

M3

ACTION A
Condition A

Required Action A.2

c. With the suppression pool water temperature > 95°F initiate pool cooling and restore the temperature to ≤ 95°F within 24 hours or be in at least the HOT SHUTDOWN CONDITION within the next 6 hours and in the COLD SHUTDOWN CONDITION within the following 30 hours

LCO 3.6.2.1.a

L1

ACTION B

Required Action B.3

When any OPERABLE ILM channel is ≤ 25/40 divisions of full scale on Range 7

L1

ACTION C

Condition C

Required Action C.1

d. With the suppression pool water temperature > 105°F during testing of ECCS or relief valves, stop all testing, initiate pool cooling and follow the action in Specification 3.7.A.1.c above.

LCO 3.6.2.1.b

LAI

ACTION A

ACTION D

Condition D

Required Action D.1

e. With the suppression pool water temperature > 110°F during the STARTUP CONDITION, HOT STANDBY CONDITION (with all control rods not inserted), or REACTOR POWER OPERATION, the reactor shall be scrammed.

LCO 3.6.2.1.c

Applicability Modes 1, 2 + 3

M1

Proposed Required Action D 2

M4

ACTION E

Condition E

M5

M4

f. With the suppression pool water temperature > 120°F following reactor isolation, depressurize to < 200 psig at normal cooldown rates.



UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

(A1)

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

(A1)

~~Specification~~

~~A. Primary Containment~~

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

Applicability

LCD 3.6.2.1

a. Minimum water level =
-6.25" (differential pressure control >0 psid)
-7.25" (0 psid differential pressure control)

b. Maximum water level =
-1"

See Justification for changes for BFN 1ST 3.5.2

See Justification for changes for BFN 3.6.2.2

(M2) Add SR 3.6.2.1.1 first frequency - once / 24 hours

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

~~Specification~~

~~A. Primary Containment~~

(A1)

~~1. Pressure Suppression Chamber~~

a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.



3.7.A. Primary Containment

3.7.A.1 (Cont'd)

Proposed Required Action A1

M3

c. With the suppression pool water temperature > 95°F initiate pool cooling and restore the temperature to ≤ 95°F within 24 hours or, be in at least the HOT SHUTDOWN CONDITION within the next 6 hours and in the COLD SHUTDOWN CONDITION within the following 30 hours.

LCO 3.6.2.1.a

L1

When any OPERABLE IRM channel is ≤ 25/40 division of full scale on Range 7

Action A Condition A

Required Action A2

L1

Action B

A1

Required Action D.3

d. With the suppression pool water temperature > 105°F during testing of ECCS or relief valves, stop all testing, initiate pool cooling and follow the action in Specification 3.7.A.1.c above.

LCO 3.6.2.1.b

LAI

Action C Condition C

Required Action C.1

Action A

Action D

e. With the suppression pool water temperature > 110°F during the STARTUP CONDITION, HOT STANDBY CONDITION (with all control rods not inserted), or REACTOR POWER OPERATION the reactor shall be scrammed.

LCO 3.6.2.1.c

Applicability MODE 1, 2+3

M1

Required Action D.1

Proposed Required Action D.2

M4

f. With the suppression pool water temperature > 120°F following reactor isolation, depressurize to < 200 psig at normal cooldown rates.

Condition E

Action E

M5

M4



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.1 - SUPPRESSION POOL AVERAGE TEMPERATURE

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 Existing Specification 3.7.A.1.e modifies the applicability governing suppression pool temperature such that the temperature limit applies only during the STARTUP CONDITION, HOT STANDBY CONDITION (with all control rods inserted), or REACTOR POWER OPERATION. Proposed LCO 3.6.2.1, Suppression Pool Average Temperature, ACTION D is applicable in Modes 1, 2, and 3. Therefore, this change is more restrictive.
- M2 CTS Surveillance Requirement 4.7.A.1.a only requires continual suppression pool temperature monitoring and logging whenever heat is added to the suppression pool during testing. Proposed SR 3.6.2.1.1 is more restrictive since it also requires this verification be performed once every 24 hours in the absence of testing.
- M3 A new Required Action has been added (proposed Required Action A.1) to verify temperature is $\leq 110^{\circ}\text{F}$ every hour, anytime temperature has exceeded 95°F . This is an additional restriction on plant operation.
- M4 When temperature exceeds 110°F , the current requirements only require the reactor to be scrammed. Proposed Required Action D.2 requires the temperature to be verified $\leq 120^{\circ}\text{F}$ every 30 minutes and a cooldown to MODE 4 within 36 hours, respectively. If temperature exceeds 120°F , the current requirements only require the RPV to be depressurized to < 200 psig at normal cooldown rates. Proposed ACTION E now requires the 200 psig limit to be attained in 12 hours, and to continue cooling down the plant to cold shutdown (MODE 4) within 36 hours. These are additional restrictions on plant operation.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.1 - SUPPRESSION POOL AVERAGE TEMPERATURE

- M5 The proposed ACTION (ACTION E) when pool temperature exceeds 120°F does not depend upon whether the reactor is isolated. If pool temperature reaches 120°F, regardless of whether the reactor is isolated, significant heat could still be added to the suppression pool and the Required Action is appropriate. Even with the reactor not isolated, there may be no heat rejection from the containment, as in the case of loss of condenser vacuum. Applying the actions regardless of whether the reactor is isolated does not introduce any operation which is unanalyzed. This change is more restrictive on plant operations.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 Details of how to reduce suppression pool temperature to within the limits have been relocated to plant procedures. Methods for restoring pool temperature are more appropriately located in plant procedures. Changes to the procedure will be controlled by the licensee controlled programs.

"Specific"

- L1 The Applicability for proposed LCO 3.6.2.1, Suppression Pool Average Temperature, is Modes 1, 2, and 3. However, this Applicability is modified within LCO 3.6.2.1 so that a lower suppression pool temperature limit applies if any Operable IRM channel is on Range 7 or above. This limit was selected so that the suppression pool temperature limits are applicable when the reactor is critical with reactor power approximately at the point of adding heat. As a result of this qualification to the Applicability statement, suppression pool temperature is required to be maintained at a temperature of less than 95°F (or less than 105°F while performing tests that add heat to the suppression pool) only when the reactor is critical with reactor power at the approximate level where heat generated is approximately equal to normal system heat losses. If the reactor is not critical or at a power below the point of adding heat, the suppression pool may be maintained at an average temperature up to 110°F. This change is less restrictive because CTS 3.7.A.1. required the lower suppression pool temperature to be less than 95°F (or less than 105°F while performing tests that add heat to the suppression pool) even if the reactor is not critical or not above the point of adding heat. If the reactor is not critical or the reactor is below the point of adding heat, there is significantly less heat generation from decay heat than assumed in the design basis. The suppression pool is designed to absorb the decay heat and sensible energy released during a reactor blowdown via safety/relief valves or from design basis accidents when the reactor has been operating continuously at full power for a



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.1 - SUPPRESSION POOL AVERAGE TEMPERATURE

considerable period of time. Any event initiated with reactor power or reactor power history less than these conditions will place considerably less heat load on the suppression pool than a DBA LOCA. This change is consistent with NUREG-1433. In addition, the shutdown requirements, if the temperature is not restored, have been modified to only require reducing power to below IRM Range 7 within 12 hours, consistent with the new Applicability.

UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

(A1)

~~Specification~~

A. Primary Containment

1. Applicability At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

LCO 3.6.2.2

- a. Minimum water level =
 - 6.25" (differential pressure control >0 psid)
 - 7.25" (0 psid differential pressure control)

- b. Maximum water level = -1"

(L1) Proposed ACTION A

(L2) Proposed ACTION B

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

~~Specification~~

~~A. Primary Containment~~

1. Pressure Suppression Chamber
SR 3.6.2.2.1

a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

[See Justification for Changes for BFN ISTS 3.5.2]

[See Justification for Changes for BFN ISTS 3.6.2.1]



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



See Justification for Changes for BFN ISTS 3.6.2.1

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

Applicability

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure

or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

- a. Minimum water level = -6.25" (differential pressure control >0 psid)
-7.25" (0 psid differential pressure control)
- b. Maximum water level = -1"

(L1) Proposed ACTION A

(L2) Proposed ACTION B

LC 3.6.2.2

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

(A1)

1. Pressure Suppression Chamber

SR 3.6.2.2.1

- a. The suppression chamber water level be checked once per day.

Whenever heat is added to the suppression pool by testing of the ECOS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.5.2

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.2.1

UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

3.7/4.7 CONTAINMENT SYSTEMS

See Justification for changes for BFN ISTS 3.6.2.1

Specification 3.6.2.2

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

(A1)

A. Primary Containment

Applicability

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

- a. Minimum water level = -6.25" (differential pressure control >0 psid)
-7.25" (0 psid differential pressure control)
- b. Maximum water level = -1"

LCO 3.6.2.2

- (L1) Proposed Action A
- (L2) Proposed Action B

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

1. Pressure Suppression Chamber

SR 3.6.2.2.1

- a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

See Justification for changes for BFN ISTS 3.5.2

See Justification for changes for BFN ISTS 3.6.2.1

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.2 - SUPPRESSION POOL WATER LEVEL

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Specific"

L1 The existing Action for suppression pool water level outside limits (Specification 3.7.A.1) allows no time to restore level. An unanticipated change in suppression pool level would require addressing the cause and aligning the appropriate system to raise or lower the pool level. These activities require some time to accomplish without undo haste. The out-of-service time is based on engineering judgement of the relative risks associated with: 1) the safety significance of the system; 2) the probability of an event requiring the safety function of the system; and 3) the relative risks associated with the plant transient and potential challenge of safety systems experienced by requiring a plant shutdown. Upon further review, and discussion with the NRC Staff, during the development of the BWR Standard Technical Specifications, NUREG 1433, a 2 hour restoration allowance was determined to be appropriate.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.2 - SUPPRESSION POOL WATER LEVEL

- L2 Per CTS, if suppression pool water level is not maintained within limits, the Specification is violated and in accordance with TS 1.0.C.1 the plant must be placed in Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours unless suppression pool water level is restored. This provides actions for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. The BFN ISTS provides Action within the Specification which could be considered less restrictive than CTS. Action B allows 12 hours to be in MODE 3 (Hot Shutdown) and 36 hours to be in MODE 4 (Cold Shutdown). The proposed Action is considered less restrictive since 12 hours is allowed to place the unit in Hot Shutdown versus the 6 hours allowed to place the unit in Hot Standby per CTS.



UNIT 1

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



(A1)

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~
LCO 3.6.2.3

1. The RHRS shall be OPERABLE #:

Applicability

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

See Justification Changes for BFN 15TS 3.5.1

~~4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Test Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



AUG 02 1989

~~LIMITING CONDITION FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

ACTION A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

(A1) ~~5. No additional surveillance required.~~

See Justification for Changes for BFN 1STS 3.8.1

ACTION B

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

(A1) ~~6. No additional surveillance required.~~

Add SR 3.6.2.3.1
SR 3.6.2.3.2

MI

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

(A1) ~~7. No additional surveillance required.~~

See Justification For Changes For BFN 1STS 3.6.2.4 + 3.6.2.5

Proposed ACTION C

(L)



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

3.5.B ~~Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

4.5.B ~~Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

~~8. No additional surveillance required.~~

ACTION D

HOT SHUTDOWN CONDITIONS in 12 hours and

M2

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within ²⁴~~36~~ hours. (L2)

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. No additional surveillance required.

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification For Changes for BFN 15 TS 3.5.1 + 3.5.2

UNIT 2

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

LCO 3.6.2.3

1. The RHRS shall be OPERABLE #:

Applicability

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

See Justification for Changes for BFN ISTS 3.5.1

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Testable Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

~~3.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

ACTION A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

(A1)

~~5. No additional surveillance required.~~

See Justification For Changes for BFN ISTS 3.8.1

ACTION B

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

(A1)

~~6. No additional surveillance required.~~

Add SR 3.6.2.3.1 MI
SR 3.6.2.3.2

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

(A1)

~~7. No additional surveillance required.~~

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.2.4 AND 3.6.2.5

(L1)

Proposed ACTION C



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

3.5.B ~~Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

4.5.B ~~Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

~~8. No additional surveillance required.~~

ACTION D

(M2)

HOT SHUTDOWN CONDITION in 12 hrs. and

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within ~~24~~³⁶ hours.

(L2)

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.5.1 + 3.5.2

UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

LCO 3.6.2.3

1. The RHRS shall be OPERABLE #:

Applicability

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

See Justification for changes for BFN ISTS 3.5.1

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle.
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Testable Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

~~LIMITING CONDITION FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

Action A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

(A1) ~~5. No additional surveillance required.~~

See Justification for Changes for BFN ISTS 3.8.1

Action B

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

(A1) ~~6. No additional surveillance required.~~

Add SR 3.6.2.3.1 (m) SR 3.6.2.3.2

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

(A1) ~~7. No additional surveillance required.~~

See Justification for Changes for BFN ISTS 3.6.2.4 And 3.6.2.5

(L1)

Proposed Action C

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.5.B ~~Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~8. No additional surveillance required.~~

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours. (L2)

Action D

(M2)

HOT SHUTDOWN CONDITION IN 12 hrs

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The B and D RHR pumps on unit 2 which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

see justification for changes for BFN ISTS 3.5.1 + 3.5.2

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.3 - RHR SUPPRESSION POOL COOLING

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 Surveillance Requirements (SR 3.6.2.3.1 and 3.6.2.3.2) have been added to ensure that the correct valve lineup for the RHR suppression pool cooling subsystems is maintained and RHR pump testing is performed to ensure the RHR suppression pool cooling subsystems remain capable of providing the overall DBA suppression pool cooling requirement. This change is consistent with NUREG-1433.
- M2 CTS 3.5.B.8 requires an orderly shutdown be initiated and the reactor to be in the Cold Shutdown Condition within 24 hours when required RHR suppression pool cooling subsystems are inoperable. Proposed Action D will require the plant be in MODE 3 (Hot Shutdown Condition) in 12 hours and MODE 4 (Cold Shutdown Condition) in 36 hours. The addition of this intermediate step to the Cold Shutdown Condition is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.3 - RHR SUPPRESSION POOL COOLING

TECHNICAL CHANGES - LESS RESTRICTIVE

- L1 Proposed ACTION C will allow 8 hours to restore required RHR suppression pool cooling subsystems to operable status prior to initiating a shutdown. The proposed 8 hour Completion Time provides some time to restore the required subsystems to Operable status, yet is short enough that operating an additional 8 hours is not risk significant. Only 8 hours is allowed since there is a substantial loss of the primary containment bypass leakage mitigation function. The 8 hour restoration time is considered acceptable due to the low probability of a DBA and because alternative methods to remove decay heat from the primary containment are still available. In addition, if the required subsystem(s) are restored to Operable status prior to the expiration of the 8 hours, a unit shutdown is averted. Thus, the potential of a unit scram occurring while shutting the unit down, which then could result in a need for a subsystem when it is inoperable, has been decreased.
- L2 The time to reach MODE 4, Cold Shutdown, has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added. These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

LCo 3.6.2.4

1. The RHRS shall be OPERABLE #:

Applicability

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

See Justification Changes for BFN ISTS 3.5.1

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Test Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



AUG 02 1989

~~LIMITING CONDITIONS FOR OPERATION~~

(A)

~~SURVEILLANCE REQUIREMENT~~

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

- 2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps—containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.
- 3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.
- 4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

4.5.B.1 (cont'd)

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 gpm against an indicated system pressure of 250 psig.

- 2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

(LAI)

See Justification for Changes for BFN ISTS 3.6.2.5

- 3. No additional surveillance required.
- 4. No additional surveillance required.

See Justification for Changes for BFN ISTS 3.5.1

AUG 02 1989

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITION FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

ACTION A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

ACTION B

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

(L1) Proposed ACTION C →

~~4.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~5. No additional surveillance required.~~

See Justification for Changes for BFN 1STS 3.8.1

(A1)

~~6. No additional surveillance required.~~

Add SR 3.6.2.4.1 (M1)

(A1)

~~7. No additional surveillance required.~~

See Justification for Changes for BFN 1STS 3.6.2.3 + 3.6.2.5



(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

ACTION D.

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within ~~24~~ hours.

36 L2

(A1)

~~8. No additional surveillance required.~~

M2

In the HOT SHUTDOWN CONDITION in 12 hours and

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification for Changes for BFN ISTS 3.5.1 + 3.5.2

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

3.5.B ~~Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(AI)

LCO 3.6.2.4

1. The RHRS shall be OPERABLE #:

applicability

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

See Justification for Change for BFN ISTS 3.5.1

4.5.B ~~Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Testable Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



AUG 02 1989

~~LIMITING CONDITIONS FOR OPERATION~~

(AI)

~~SURVEILLANCE REQUIREMENT~~

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

- 2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.
- 3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.
- 4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

4.5.B.1 (cont'd)

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 gpm against an indicated system pressure of 250 psig.

2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

SR 36.2.4.2

(LAI)

See Justification for Changes for BFN ISTS 3.6.2.5

3. No additional surveillance required.

4. No additional surveillance required.

See Justification for Changes for BFN ISTS 3.5.1

~~LIMITING CONDITION FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ AUG 02 1989

~~3.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

ACTION A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

~~AI 5. No additional surveillance required.~~

See Justification for Changes for BFN ISTS 3.8.1

ACTION B

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

~~AI 6. No additional surveillance required.~~

Add. SR 3.6.2.4.1 MI

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

~~AI 7. No additional surveillance required.~~

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.2.3 + 3.6.2.5

LI Proposed ACTION C

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

ACTION D

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

~~(A1) No additional surveillance required.~~

in the HOT SHUTDOWN CONDITION in 12 hours and

(M2)

(L2)

(36)

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

(See Justification for Changes for BFN ISTS 3.5.1 + 3.5.2)



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



~~LIMITING CONDITIONS FOR OPERATION~~

SURVEILLANCE REQUIREMENTS

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

LCO 3.6.2.4

1. The RHRS shall be OPERABLE #:

Applicability

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

See Justification for BFN 15TS 3.5.1

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Testable Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

(A1)

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

- 2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.
- 3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.
- 4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

~~4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

4.5.B.1 (cont'd)

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 gpm against an indicated system pressure of 250 psig.

- 2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

SR 3.6.2.4.2

(LAI)

See Justification for changes for BFN ISTS 3.6.2.5

- 3. No additional surveillance required.
- 4. No additional surveillance required.

See Justification for changes for BFN ISTS 3.5.1



LIMITING CONDITION FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.5 B. ~~Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

Action A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

Action B

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

(2) Proposed Action C

4.5 B. ~~Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

5. ~~No additional surveillance required.~~ (A1)

See Justification for Changes for BFN ISTS 3.8.1

(A1) ~~6. No additional surveillance required.~~

Add SR 3.6.2.4.1 (m1)

(A1) ~~7. No additional surveillance required.~~

See Justification for Changes for BFN ISTS 3.6.2.3 & 3.6.2.5



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.5.B ~~Residual Heat Removal System (RHRG) (LPCI and Containment Cooling)~~

4.5.B ~~Residual Heat Removal System (RHRG) (LPCI and Containment Cooling)~~

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within ~~24~~ hours.

~~8. No additional surveillance required.~~

Action D

In the HOT SHUTDOWN CONDITION in 12 hours AND

L2

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The B and D RHR pumps on unit 2 which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification for Changes for BFN ISTS 3.5.1 & 3.5.2

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.4 - RHR SUPPRESSION POOL SPRAY

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 Surveillance Requirement (SR 3.6.2.4.1) has been added to ensure that the correct valve lineup for the RHR suppression pool spray subsystems is maintained. This ensures that the RHR suppression pool spray subsystems remain capable of providing the overall DBA suppression pool spray requirement. This change is consistent with NUREG-1433.
- M2 CTS 3.5.B.8 requires an orderly shutdown be initiated and the reactor to be in the Cold Shutdown Condition within 24 hours when required RHR suppression pool spray subsystems are inoperable. Proposed Action D will require the plant be in MODE 3 (Hot Shutdown Condition) in 12 hours and MODE 4 (Cold Shutdown Condition) in 36 hours. The addition of this intermediate step to the Cold Shutdown Condition is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.4 - RHR SUPPRESSION POOL SPRAY

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 Details of the methods of performing surveillance test requirements have been relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 Proposed ACTION C will allow 8 hours to restore required RHR suppression pool spray subsystems to operable status prior to initiating a shutdown. The proposed 8 hour Completion Time provides some time to restore the required subsystems to Operable status, yet is short enough that operating an additional 8 hours is not risk significant. Only 8 hours is allowed since there is a substantial loss of the primary containment bypass leakage mitigation function. The 8 hour restoration time is considered acceptable due to the low probability of a DBA and because alternative methods to remove decay heat from the primary containment are still available. In addition, if the required subsystem(s) are restored to Operable status prior to the expiration of the 8 hours, a unit shutdown is averted. Thus, the potential of a unit scram occurring while shutting the unit down, which then could result in a need for a subsystem when it is inoperable, has been decreased.
- L2 The time to reach MODE 4, Cold Shutdown has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added. These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.



UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

Applicability

- 1. The RHRS shall be OPERABLE #:
- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

See Justification for Changes for BFN ISTS 3.5.1

(A1)

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Test Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



AUG 02 1989

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENT~~

3.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

- 2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps—containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.
- 3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.
- 4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

4.5.B.1 (cont'd)

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 gpm against an indicated system pressure of 250 psig.

(LAI)

SR 3.6.2.5.2

- 2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

See Justification for changes for BFN ISTS 3.6.2.4

- 3. No additional surveillance required.

- 4. No additional surveillance required.

See Justification for changes for BFN ISTS 3.5.1



AUG 02 1989

~~LIMITING CONDITION FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

ACTION A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

Proposed ACTION C

(L1)

~~4.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1) ~~5. No additional surveillance required.~~

(A1) ~~6. No additional surveillance required.~~

(M) Add SR 3.6.2.5.1

(A1) ~~7. No additional surveillance required.~~

See Justification for Changes for BFN ISTS 3.6.2.3 and 3.6.2.4

See Justification for Changes for BFN ISTS 3.8.1



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

~~4.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours. (36) (L2)

(A1) 8. No additional surveillance required. (M2)

in the HOT SHUTDOWN CONDITION in 12hrs and

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification for Change for BFN ISTS 3.5.1 & 3.5.2



UNIT 2
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A)

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

Applicability

- 1. The RHRS shall be OPERABLE #:

 - (1) PRIOR TO STARTUP from a COLD CONDITION; or
 - (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

See Justification for Changes for BFN ISTS 3.5.1

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Testable Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



AUG 02 1989

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENT

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

- 2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps—containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.

- 3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.

- 4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

4.5.B.1 (cont'd)

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 gpm against an indicated system pressure of 250 psig.

- 2. An ~~air test~~ on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

SR 3.6.2.5.2

(LAI)

See Justification for Changes for BFN 1STS 3.6.2.4

- 3. No additional surveillance required.

- 4. No additional surveillance required.

See Justification for Changes for BFN 1STS 3.5.1

LIMITING CONDITION FOR OPERATION

~~SURVEILLANCE REQUIREMENTS~~

AUG 02 1989

~~3.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

ACTION A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

~~(A1) 5. No additional surveillance required.~~

See Justification for Changes for BFN 15TS 3.8.1

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

~~(A1) 6. No additional surveillance required.~~

(M1) Add SR 3.6.2.5.1

ACTION B

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

~~(A1) 7. No additional surveillance required.~~

See Justification for Changes for BFN 15TS 3.6.2.3 and 3.6.2.4

(L1) Proposed ACTION C

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

APR 19 1994

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(A1) 8. No additional surveillance required.

in the HOT SHUTDOWN CONDITION in 12 hours and (M2)

(L2)

36

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.5.1 + 3.5.2



UNIT 3
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

Unit 3 5



(A1)

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

Applicability

- 1. The RHRS shall be OPERABLE #:
- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Testable Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

See Justification for Changes for BFN ISTS 3.5.1

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



(A1)

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

- 2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps—containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.
- 3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.
- 4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

4.5.B.1 (cont'd)

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 gpm against an indicated system pressure of 250 psig.

(A1)

SR
3.6.2.5.2

- 2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

See Justification for Changes for BFN 15TS 3.6.2.4

3. No additional surveillance required.

4. No additional surveillance required.

See Justification for Changes for BFN 15TS 3.5.1



~~LIMITING CONDITION FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 B. Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

Action A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHR (containment cooling mode) are OPERABLE.

Action B

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHR (containment cooling mode) are OPERABLE.

7. If two access paths of the RHR (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

(L1) Proposed Action C →

~~4.5 B. Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1) ~~5. No additional surveillance required.~~

See Justification for Changes for BFN ISTS 3.8.1

(A1) ~~6. No additional surveillance required.~~

(M1) Add SR 3.6.2.5.1 →

(A1) ~~7. No additional surveillance required.~~

See Justification for Changes for BFN ISTS 3.6.2.3 and 3.6.2.4



~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(A1) ~~8. No additional surveillance required.~~

(L2)

(36)

in the HOT SHUTDOWN Condition in 12 hours and

(M2)

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The B and D RHR pumps on unit 2 which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification for Changes for BFN ISTS 3.5.1 + 3.5.2



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.5 - RHR DRYWELL SPRAY**

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 Surveillance Requirement (SR 3.6.2.5.1) has been added to ensure that the correct valve lineup for the RHR drywell spray subsystems is maintained. This ensures that the RHR drywell spray subsystems remain capable of providing the overall DBA drywell spray requirement. This change is consistent with NUREG-1433.
- M2 CTS 3.5.B.8 requires an orderly shutdown be initiated and the reactor to be in the Cold Shutdown Condition within 24 hours when required RHR drywell spray subsystems are inoperable. Proposed Action D will require the plant be in MODE 3 (Hot Shutdown Condition) in 12 hours and MODE 4 (Cold Shutdown Condition) in 36 hours. The addition of this intermediate step to the Cold Shutdown Condition is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.5 - RHR DRYWELL SPRAY

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 Details of the methods of performing surveillance test requirements have been relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 Proposed ACTION C will allow 8 hours to restore required RHR drywell cooling subsystems to operable status prior to initiating a shutdown. The proposed 8 hour Completion Time provides some time to restore the required subsystems to Operable status, yet is short enough that operating an additional 8 hours is not risk significant. Only 8 hours is allowed since their loss substantially reduces the ability to maintain primary containment within design limits. The 8 hour restoration time is considered acceptable due to the low probability of a DBA and because alternative methods to remove decay heat from the primary containment are still available. In addition, if the required subsystem(s) are restored to Operable status prior to the expiration of the 8 hours, a unit shutdown is averted. Thus, the potential of a unit scram occurring while shutting the unit down, which then could result in a need for a subsystem when it is inoperable, has been decreased.
- L2 The time to reach MODE 4, Cold Shutdown has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added. These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.



UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



A1

~~LIMITING CONDITIONS FOR OPERATION~~

~~CONTAINMENT REQUIREMENTS~~

~~3.7.A. Primary Containment~~

~~6. Drywell-Suppression Chamber Differential Pressure~~

LCD
3.6.2.6

a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.1 psid except as specified in (1) and (2) below:

Applicability

(1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.1 psid 24 hours prior to a scheduled shutdown.

L1

LCD
Note

(2) This differential may be decreased to less than 1.1 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system and the drywell-pressure suppression chamber vacuum breakers.

ACTIONS
A+B

b. If the differential pressure of Specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six-hour period, an orderly shutdown shall be initiated and the reactor shall be in the ~~COLD SHUTDOWN CONDITION~~ within 24 hours.

8 hour

L2

Reduce THERMAL POWER to $\leq 15\%$ in 12 hrs

~~3.7.A. Primary Containment~~

~~6. Drywell-Suppression Chamber Differential Pressure~~

SR 3.6.2.6.1

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

A2

12 hours



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.A. Primary Containment

(A1)

6. Drywell Suppression Chamber Differential Pressure

LCO
3.6.2.6

a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.1 psid except as specified in (1) and (2) below:

Applicability

(1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.1 psid 24 hours prior to a scheduled shutdown.

(L1)

LCO
Note

(2) This differential may be decreased to less than 1.1 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system and the drywell-pressure suppression chamber vacuum breakers.

ACTIONS
A+B

b. If the differential pressure of Specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six hour period, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.
Reduce THERMAL POWER to $\leq 15\%$ in 12 hrs.

8 hour

(L2)

4.7.A. Primary Containment

6. Drywell Suppression Chamber Differential Pressure

SR 3.6.2.6.1

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

(A2)

12 hours



UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

(A1)

~~3.7.A. Primary Containment~~

~~4.7.A. Primary Containment~~

~~6. Drywell Suppression Chamber Differential Pressure~~

~~6. Drywell Suppression Chamber Differential Pressure~~

SR 3.6.2.6.1

LCO
3.6.2.6

a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.1 psid except as specified in (1) and (2) below:

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

(A2) 12 hours

Applicability

(1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.1 psid 24 hours prior to a scheduled shutdown.

(L1)

LCO Note

(2) This differential may be decreased to less than 1.1 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system and the drywell-pressure suppression chamber vacuum breakers.

Action A+B

b. If the differential pressure of Specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six-hour period, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours. Reduce thermal power to $\leq 15\%$ in 12 hours.

8 hours
(L2)

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.6
DRYWELL-TO-SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The Frequency for verifying the pressure differential between the drywell and the suppression chamber has been changed to 12 hours from shiftly. CTS Table 1.1 defines shiftly as at least once per 12 hours. As such, this is a change in presentation only and is therefore administrative.

TECHNICAL CHANGES - LESS RESTRICTIVE

- L1 The proposed change revises the required initiation point for establishing differential pressure between the drywell and suppression chamber. By increasing the initiation point following startup to 15% rated thermal power (RTP) (CTS initiation point is operating temperature and pressure, which is about 1% RTP), the drywell pressure and temperature will have sufficient time to stabilize prior to establishing the required differential pressure. As long as reactor power is below 15% RTP, the probability of an event that generates excessive loads on primary containment occurring within the first 24 hours of a startup or within the last 24 hours before shutdown is low. 24 hours is considered a reasonable amount of time to allow plant personnel to establish the required differential pressure.
- L2 CTS 3.7.A.6.b allows 6 hours to restore the differential pressure before initiating an orderly shutdown, which requires the plant to be in Cold Shutdown within 24 hours. The proposed actions allow 8 hours to restore differential pressure and 12 hours to reduce thermal power to \leq 15% RTP. Below this power level, per the proposed Specification, the LCO is no longer applicable (See Comment L1 above).

PAGE 1 OF 1

UNIT 1

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



APR 29 1991

~~3.7/4.7 CONTAINMENT SYSTEMS~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.7.F. Primary Containment Purge System

4.7.F. Primary Containment Purge System

3.7.F.3 (Continued)

- . these primary containment isolation valves is governed by Technical Specification 3.7.D.
- b. Pressure control of the containment is normally performed by VENTING through 2-inch primary containment isolation valves which route effluent to the Standby Gas Treatment System. The OPERABILITY of these primary containment isolation valves is governed by Technical Specification 3.7.D.

See Justification for Changes for CTS 3.7.F/4.7.F

~~3.7.G. Containment Atmosphere Dilution System (CAD)~~

(A1)

~~4.7.G. Containment Atmosphere Dilution System (CAD)~~

LCO 3.6.3.1

1. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE with:
 - a. Two independent systems capable of supplying nitrogen to the drywell and torus.

~~1. System Operability~~

(LA1)

- a. Cycle each solenoid operated air/nitrogen valve through at least one complete cycle of full travel in accordance with Specification 1.0.MM, and at least once per month verify that each manual valve in the flow path is open.

(M1)

SR 3.6.3.1.2

SR 3.6.3.1.1

- b. A minimum supply of 2,500 gallons of liquid nitrogen per system.

SR 3.6.3.1.1

- b. Verify that the CAD System contains a minimum supply of 2,500 gallons of liquid nitrogen twice per week.

(L1)

DEC 07 1994

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

A1

~~3.7.G. Containment Atmosphere Dilution System (CAD)~~

~~4.7.G. Containment Atmosphere Dilution System (CAD)~~

LCD 3.6.3.1 + Applicability

2. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE whenever the reactor is in the RUN MODE. or Startup Mode M2

2. When FCV 84-8B is inoperable, each solenoid operated air/nitrogen valve of System B shall be cycled through at least one complete cycle of full travel and each manual valve in the flow path of System B shall be verified open at least once per week.

A3

ACTIONS A

3. If one system is inoperable, the reactor may remain in operation for a period of 30 days provided all active components in the other system are OPERABLE.

Note to Required Action A.1 A2

ACTION B

4. If Specifications 3.7.G.1 and 3.7.G.2, or 3.7.G.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the ~~COLD SHUTDOWN CONDITION~~ within 24 hours.

MODE 3 within 12 hours M3

5. Primary containment pressure shall be limited to a maximum of 30 psig during repressurization following a loss of coolant accident.

LA 2

A3

6. System A may be considered OPERABLE with FCV 84-8B inoperable provided that all active components in System B and all other active components in System A are OPERABLE.
7. Specifications 3.7.G.6 and 4.7.G.2 are in effect until the first Cold Shutdown of unit 1 after July 20, 1984 or until January 17, 1985 whichever occurs first.



UNIT 2

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.7.F. Primary Containment Purge System

3.7.F.3 (Continued)

these primary containment isolation valves is governed by Technical Specification 3.7.D.

- b. Pressure control of the containment is normally performed by VENTING through 2-inch primary containment isolation valves which route effluent to the Standby Gas Treatment System. The OPERABILITY of these primary containment isolation valves is governed by Technical Specification 3.7.D.

4.7.F. Primary Containment Purge System

See Justification for Changes for CTS 3.7.F/4.7.F

3.7.G. Containment Atmosphere Dilution System (CAD)

(A1)

LCO 3.6.3.1 1. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE with:

- a. Two independent systems capable of supplying nitrogen to the drywell and torus.

- SR 3.6.3.1.1 b. A minimum supply of 2,500 gallons of liquid nitrogen per system.

4.7.G. Containment Atmosphere Dilution System (CAD)

1. System Operability

(LAI)

a. Cycle each solenoid operated air/nitrogen valve through at least one complete cycle of full travel in accordance with Specification 1.0.MM.

(M1)

SR 3.6.3.1.2

and at least once per month verify that each manual valve in the flow path is open.

SR 3.6.3.1.1

b. Verify that the CAD System contains a minimum supply of 2,500 gallons of liquid nitrogen twice per week.

(L1)



DEC 07 1994

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.G. Containment Atmosphere Dilution System (CAD)~~

~~4.7.G. Containment Atmosphere Dilution System (CAD)~~

LCO 3.6.3.1
Applicability

2. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE whenever the reactor is in the RUN MODE or STARTUP MODE

(A1)

(M2)

ACTION A

3. If one system is inoperable, the reactor may remain in operation for a period of 30 days provided all active components in the other system are OPERABLE.

Note to Required Action A.1 (A2)

ACTION B

4. If Specifications 3.7.G.1 and 3.7.G.2, or 3.7.G.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

Mode 3 within 12 hours (M3)

5. Primary containment pressure shall be limited to a maximum of 30 psig during repressurization following a loss of coolant accident.

(LA2)



UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

3.7.F. Primary Containment Purge System

4.7.F. Primary Containment Purge System

3.7.F.3 (Continued)

these primary containment isolation valves is governed by Technical Specification 3.7.D.

- b. Pressure control of the containment is normally performed by VENTING through 2-inch primary containment isolation valves which route effluent to the Standby Gas Treatment System. The OPERABILITY of these primary containment isolation valves is governed by Technical Specification 3.7.D.

See Justification for Changes for CTS 3.7.F/4.7.F

~~3.7.G. Containment Atmosphere Dilution System (CAD)~~ (A1)

~~4.7.G. Containment Atmosphere Dilution System (CAD)~~

LLD 3.6.3.1

1. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE with:

- a. Two independent systems capable of supplying nitrogen to the drywell and torus.

SR 3.6.3.1.1

- b. A minimum supply of 2,500 gallons of liquid nitrogen per system.

~~1. System Operability~~

(LA1)

- a. Cycle each solenoid operated air/nitrogen valve through at least one complete cycle of full travel in accordance with Specification 1.0.MM, and at least once per month verify that each manual valve in the flow path is open.

(M1)

SR 3.6.3.1.2

SR 3.6.3.1.1

- b. Verify that the CAD System contains a minimum supply of 2,500 gallons of liquid nitrogen twice per week.

(L1)



~~3.7.G. Containment Atmosphere Dilution System (CAD)~~

(A1)

~~4.7.G. Containment Atmosphere Dilution System (CAD)~~

2. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE whenever the reactor is in the RUN MODE, A or STARTUP MODE (M2)

LCO 3.6.3.1
+
Applicability

(A2)

Note to Required Action A.1

3. If one system is inoperable, the reactor may remain in operation for a period of 30 days provided all active components in the other system are OPERABLE.

Action A

4. If Specifications 3.7.G.1 and 3.7.G.2, or 3.7.G.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the ~~COLD SHUTDOWN CONDITION~~ within 24 hours.

Action B

(M3)

MODE 3
within 12 hours

5. Primary containment pressure shall be limited to a maximum of 30 psig during repressurization following a loss of coolant accident.

(LA2)

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.3.1 - CONTAINMENT AIR DILUTION SYSTEM

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

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A2 A NOTE was added specifying LCO 3.0.4 is not applicable. Since the current Technical Specifications do not have LCO 3.0.4, stating it is not applicable constitutes an administrative change.

A3 Unit 1 CTS 3.7.G.6 & 7 and 4.7.G.2 have been deleted. These Specifications were special provisions that expired January 17, 1985, and therefore, no longer apply. As such, the proposed deletion is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 The Surveillance Requirement has been revised to include each manual, power operated, and automatic valve that is not locked, sealed, or otherwise secured in position.

M2 This change adds MODE 2 (STARTUP MODE) to the Applicability to go along with MODE 1 (RUN MODE) which is already required. The CAD System is required to maintain the oxygen concentration in the primary containment below the flammability limit following a LOCA. Adding a new MODE to the Applicability constitutes a more restrictive change. This change is consistent with NUREG-1433.



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.3.1 - CONTAINMENT AIR DILUTION SYSTEM**

- M3 Proposed ACTION C is more restrictive since it requires the unit to be placed in MODE 3 in 12 hours versus CTS 3.7.G.5 which requires that an orderly shutdown be initiated and the reactor to be in the COLD SHUTDOWN CONDITION within 24 hours. In addition, since the existing Specification (CTS 3.7.G.2) is only applicable during the RUN mode (MODE 1), failure to meet the existing specification would only require the unit be placed in at least STARTUP/HOT STANDBY (MODE 2) in 24 hours since at that time CTS 3.7.G.2 is again met.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 This Surveillance is being relocated to plant procedures (IST program) since these valves are tested as part of the IST program. As such, it is not needed to be specified as a specific Surveillance Requirement. If during testing or routine use of the system they are found to be inoperable, the appropriate ACTIONS would be taken. This change is consistent with the BWR Standard Technical Specifications, NUREG 1433.
- LA2 This requirement has been relocated to plant procedures. This type of action is a post-accident action routinely governed by the emergency operating procedures. Any changes to the procedures would be controlled by the licensee controlled programs.

"Specific"

- L1 The Frequency of this Surveillance has been extended to 31 days, similar to other surveillances on tank content (e.g., diesel fuel oil). The nitrogen tank contents only decrease when nitrogen is being added to the drywell, and this evolution is a manually actuated and secured evolution (i.e., it is a very controlled evolution). If nitrogen was being added, it would be monitored more closely. Thus, since there are very positive means to ensure nitrogen tank volume is monitored if being used, and volume does not decrease due to "automatic, unmonitored" use, the 31 day Frequency is considered appropriate.

~~3.7.4.7 CONTAINMENT SYSTEMS~~

Specification 3.6.3.2

~~LIMITING CONDITIONS FOR OPERATION~~

~~STAFF-LINE REQUIREMENTS~~

(A1)

~~3.7.A Primary Containment~~

~~4.7.A Primary Containment~~

3.7.A.4 (Cont'd)

4.7.A.4 (Cont'd)

(See Justification for Change For BFN UTS 3.6.17)

- c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.
- d. If Specifications 3.7.A.4.a, 3.7.A.4.b, or 3.7.A.4.c cannot be met, the unit shall be placed in a COLD SHUTDOWN CONDITION in an orderly manner within 24 hours.

- c. Each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.22.
- d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

(A1) ~~5 Oxygen Concentration~~

LC0 3.6.3.2.a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

~~5 Oxygen Concentration~~
SR 3.6.3.2.1

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

Applicability

b. Within the 24-hour period subsequent to placing the reactor in the RUN MODE following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a COLD SHUTDOWN CONDITION within 24 hours.

c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

d. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN CONDITION within 24 hours.

Proposed ACTION A

ACTION B

BFN Unit 1

3.7/4.7-11

AMENDMENT NO 159

PAGE 1 OF 1

(M1)

~~3.7/A.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

(A1)

~~A.7.A Primary Containment~~

3.7.A.4 (Cont'd)

4.7.A.4 (Cont'd)

- c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.
- d. If Specifications 3.7.A.4.a, .b, or .c cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.

- c. Each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.MM.
- d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.1.7

(A1) ~~5. Oxygen Concentration~~

~~5. Oxygen Concentration~~

SR3.6.3.2.1

- LC03.6.3.2 a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen/gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

- a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

Applicability

- b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

- b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

- c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a Cold Shutdown condition within 24 hours.

- c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

- d. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

Propose ACTION A (L2)

ACTION B

BFN Unit 2

3.7/4.7-11

(M1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

(A1)

~~4.7.A Primary Containment~~

3.7.A.4 (Cont'd)

- c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.
- d. If Specifications 3.7.A.4.a, 3.7.A.4.b, or 3.7.A.4.c, cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.

See Justification for Changes for BFNISTS 3.6.1.7

4.7.A.4 (Cont'd)

- c. Once each operating cycle, each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.MM.
- d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

~~(A1) 5. Oxygen Concentration~~

LCO 3.6.3.2

- a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

(ZA1)

(A2)

Applicability

- b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

(L1)

- c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a Cold Shutdown condition within 24 hours.

(LA3)

- d. If the specifications of 3.7.A.5.a through 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

Action B

(M1)

~~5. Oxygen Concentration~~

SR 3.6.3.2.1

- a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

(L3)

(LA4)

(LA2)

- b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

(LA3)

- c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

Proposed ACTION A (AL2)

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.3.2
PRIMARY CONTAINMENT OXYGEN CONCENTRATION

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 This statement has been deleted since it is unnecessary. With the reactor in power operation, reactor coolant pressure will always be above 100 psig.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 The requirement to place the plant in Cold Shutdown condition within 24 hours when the limit is not restored within the required Completion Time is revised to reflect placing the plant in a non-applicable condition. CTS 1.0.C.1 states action requirements are applicable during the operational conditions of each specification. Therefore, the requirement to place the plant in Cold Shutdown is not applicable if thermal power is reduced to < 15% RTP (outside the applicable condition) within 8 hours. The current action allows 24 hours to place the plant in a non-applicable condition. As such, this is an additional restriction on plant operation.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 The details of how to reduce oxygen concentration to less than 4% have been eliminated from the ISTS. This type of detail will be retained in plant procedures and/or system operating instructions.
- LA2 Details on the methods of performing surveillances has been relocated to

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.3.2
PRIMARY CONTAINMENT OXYGEN CONCENTRATION

plant procedures. Changes to plant procedures will be controlled by the licensee controlled programs.

- LA3 Requirements for controlling the use of plant control air to supply the pneumatic control system inside the primary containment and the associated surveillance have been relocated to the Technical Requirements Manual (TRM).
- LA4 The requirement to record the containment oxygen concentration will be relocated to plant procedures. Changes to plant procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 The 24 hour allowance for inerting on startup has been changed to allow 24 hours after exceeding 15% power instead of the current Run Mode requirement (approximately 5%). The 24 hour allowance for de-inerting on shutdown has been changed to allow 24 hours prior to reducing below 15% power. These small differences provide some added time to inert or de-inert the drywell, and provide consistency with BWR Standard Technical Specifications, NUREG-1433. These minor changes are justified, since the time allowed without an inerted drywell is only increased slightly, and the fact that at low power levels, hydrogen generation is very small compared to higher power levels.
- L2 Currently, no time is provided to restore oxygen concentration to within limit prior to requiring a plant shutdown. Proposed Required Action A.1 and associated Completion Time will allow 24 hours to restore oxygen to within the limit prior to requiring a plant shutdown. During this time, the CAD System is normally still OPERABLE, thus a means to prevent combustible mixtures still exists. This new ACTION would possibly prevent unnecessary shutdown and the increased potential for transients associated with the shutdown.
- L3 The periodic verification of oxygen concentration in the primary containment has been changed from a daily verification to a weekly verification. The primary containment is inerted to maintain oxygen concentrations within limits. The primary containment leak rate is established for each operating cycle and any changes during normal operation usually occur very slowly. Other changes to primary containment integrity, such as PCIV operability problems, are indicated by other means to the plant operator and appropriate actions are contained in other technical specifications.

UNIT 1

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



MAR 30 1990

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.7.B. Standby Gas Treatment System

3.7.B.4 (Cont'd)

- b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

3.7.C. Secondary Containment

(A2)

- LCD 3.6.4.1 1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.
- Applicability

OPERABLE

CONDITION A+C

(A2)

operable

- 2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:

Proposed Note to Required Action C.1

- a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel. *Immediately* (M4)

ACTION C

ACTIONS A+B

- b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

4.7.B. Standby Gas Treatment System

See Justification for Changes for BFN 15TS 3.6.4.7

(A1)

4.7.C. Secondary Containment

- 1. Secondary containment surveillance shall be performed as indicated below:

SR 3.6.4.1.3 + SR 3.6.4.1.4

- a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system leakage rate of not more than 12,000 cfm shall be demonstrated at each refueling outage prior to refueling.

(LA1)

(M2)

(L1)

Within 120 Seconds

(M1)

4 hour

- 2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

(LA2)

(M3)

Add SRs 3.6.4.1.1 and 3.6.4.1.2



~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.G. Secondary Containment~~

- 3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
- 4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

M5

See Justification for Changes for BFN ISTS 3.6.1.3.

D. Primary Containment Isolation Valves

- 1. When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE* except as specified in 3.7.D.2.

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

D. Primary Containment Isolation Valves

- 1. The primary containment isolation valves surveillance shall be performed as follows:
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation



UNIT 2

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



MAR 30 1990

~~3.7/4.7 CONTAINMENT SYSTEMS~~

LIMITING CONDITIONS FOR OPERATION

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.7.B. Standby Gas Treatment System

3.7.B.4 (Cont'd)

b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

4.7.B. Standby Gas Treatment System

See Justification for Changes for BFN 1STS 3.6.4.3

3.7.C. Secondary Containment

LC0 3.6.4.1 Applicability

1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

(A2) OPERABLE

4.7.C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

SR 3.6.4.1.3 + SR 3.6.4.1.4

2. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system leakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

(LA1) within 120 seconds
(M2) (M1) for 1 hour
(L1) (LA2)

CONDITION A+C

(A2) operable

2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:

(L2)

Proposed Note to Required Action C.1

ACTION C

a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel immediately (M4)

ACTIONS A+B

b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

(M3) Add SRs 3.6.4.1.1 and 3.6.4.1.2



NOV 16 1992

3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.7.6. Secondary Containment

- 3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
- 4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

(M5)

See Justification for Changes for BFN 1STS 3.6.1.3

D. Primary Containment Isolation Valves

- 1. When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE* except as specified in 3.7.D.2.

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

D. Primary Containment Isolation Valves

- 1. The primary containment isolation valves surveillance shall be performed as follows:
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.B. Standby Gas Treatment System

4.7.B. Standby Gas Treatment System

3.7.B.4 (Cont'd)

- b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

See Justification For Changes For GFN 1ST 3.6.4.3

3.7.C. Secondary Containment

4.7.C. Secondary Containment

- 1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

LCO 3.6.4.1 Applicability

* LCO not applicable until just prior to loading fuel into the Unit 3 reactor vessel, provided the Unit 3 reactor zone is not required for secondary containment integrity for other units.

OPERABLE

A3

A2

Condition A+C

Operable

- 2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:

L2

Proposed Note to Revise Action C.1.

- a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel.

Action C

Immediately M4

- b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

Actions A+B

- 1. Secondary containment surveillance shall be performed as indicated below:

SR 3.6.4.1.3 & SR 3.6.4.1.4

- a. Secondary containment capability to maintain 1/4 inch of water vacuum

LA1

under calm wind (< 5 mph) conditions within 120 seconds

M2

with a system leakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

L1

LA2

per 1 hour M1

- 2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

M3

Add SRs 3.6.4.1.1 and 3.6.4.1.2



~~3.7.C. Secondary Containment~~

- 3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
- 4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

(M5)

See Justification for Changes for BFN ISTS 3.6.1.3

D. Primary Containment Isolation Valves

- 1. When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE* except as specified in 3.7.D.2.

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

D. Primary Containment Isolation Valves

- 1. The primary containment isolation valves surveillance shall be performed as follows:
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.1 - SECONDARY CONTAINMENT**

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The definition of SECONDARY CONTAINMENT INTEGRITY has been deleted from the proposed Technical Specifications. In its place the requirement for secondary containment is that it "shall be OPERABLE." This was done because of the confusion associated with these definitions compared to its use in the respective LCO. The change is editorial in that all the requirements are specifically addressed in the proposed LCO for the secondary containment and in the Secondary Containment Isolation Valves and Standby Gas Treatment System Specifications. The Applicability has been reworded to be consistent with the new definitions of MODES and to have a positive statement as to when it is applicable, not when it is not applicable. Therefore the change is purely a presentation preference adopted by the BWR Standard Technical Specifications, NUREG-1433.

- A3 Amendment 159 to Unit 3 Technical Specifications added a provision to allow separating the Unit 3 reactor zone from the secondary containment envelope under certain conditions (prior to fuel loading) to expedite Unit 3 constructions activities during Unit 2 operation. This provision is no longer needed and can no longer be applied. Therefore the * Note to TS 3.7.C.1 has been deleted. This change is considered administrative since it deletes a requirement that no longer applies.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 This Surveillance (it appears to be only one Surveillance, though it is in two parts) has been broken into two separate Surveillances, SR



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.1 - SECONDARY CONTAINMENT

3.6.1.4.3 and SR 3.6.1.4.4. The tests will ensure the ability of the secondary containment to maintain 1/4 inch vacuum, and in addition, SR 3.6.4.1.3 will ensure the vacuum is attained in 120 seconds, while SR 3.6.4.1.4 will ensure it maintains the vacuum for 1 hour. These new requirements are additional restrictions on plant operation.

- M2 The analysis for secondary containment drawdown assumes two SGT subsystems are needed. Thus, the test now specifies the minimum number of operating SGT subsystems and the total flow rate. To ensure all three SGT subsystems are tested (since the test does not specify that all SGT subsystems must be tested) the Frequency is on a STAGGERED TEST BASIS, which will ensure all three SGT subsystems are tested in 2 cycles. These are additional restrictions of plant operation.
- M3 Two new Surveillance Requirements have been added. SR 3.6.4.1.1 will verify that all secondary containment hatches are closed and sealed every 31 days. SR 3.6.4.1.2 will verify that each access door is closed, except when used for opening, and then one door is closed, every 31 days. These are additional restrictions on plant operation.
- M4 This change requires the movement of irradiated fuel in secondary containment and CORE ALTERATIONS to be "Immediately" suspended if secondary containment is inoperable. In addition, action must be "Immediately" initiated to suspend operations with the potential to drain the reactor vessel in this Condition. The current specification does not establish a time limit to suspend these activities. Immediately suspending these activities minimizes the probability of a fission product release if a reactivity event occurs while the secondary containment is inoperable. Also, immediately initiating action to suspend operation with the potential to drain the reactor vessel will minimize the potential for reactor vessel draindown and subsequent potential for fission release. Imposing a time limit to suspended these activities is a more restrictive change.
- M5 The reactor building is divided into four ventilation zones which may be isolated independently of each other. The refueling room which is common to all three units forms the refueling zone. The individual units below the refueling floor form the other three reactor zones. The zone system is not an engineered safeguard, and the failure of the zone system would not in any way prevent isolation or reduce the capacity of the Secondary Containment System. If the internal zone boundaries should fail, the entire reactor building still meets the requirements of secondary containment. CTS 3.7.C requires the secondary containment integrity to be maintained in the reactor zone and refueling zone at all times except as specified in 3.7.C.2 and 3.7.C.4 respectively. If secondary containment cannot be maintained in the reactor zone, fuel



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.1 - SECONDARY CONTAINMENT

secondary containment must be restored within 4 hours or all reactor shall be shut down. If secondary containment cannot be maintained in the refueling zone, the handling of spent fuel and all operations over spent fuel pools and open reactor wells shall be prohibited.

Currently, a combined secondary containment integrity test is performed to demonstrate Technical Specification operability. In addition, due to leakage between zones, zone integrity is difficult to maintain. As such, secondary containment integrity is maintained on the three reactor zones and the refueling zone at all time. Therefore, the separate Specification that only prohibits the handling of spent fuel and all operations over spent fuel pools and open reactor wells when refueling zone integrity is not maintained is not necessary and has been deleted

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 This design detail/requirement has been relocated to the Background section of the Bases for ITS 3.6.4.3, "Standby Gas Treatment System," and to plant procedures governing this Surveillance Requirement. Any changes to this requirement will require a licensee controlled program evaluation.
- LA2 The requirement to operate the Standby Gas Treatment System after a secondary containment violation is determined and has been isolated (i.e., restored) to check if it can maintain the proper vacuum is being relocated to plant procedures. Any time the OPERABILITY of a system or component has been affected by maintenance, replacement, or repair, post maintenance testing is required to demonstrate OPERABILITY of the system or components. Explicit post maintenance surveillance testing has therefore been deleted from the Technical Specifications and will be relocated to the appropriate plant procedures. Any changes to the requirement will require a licensee controlled program evaluation. This change is consistent with NUREG-1433.

"Specific"

- L1 The proposed surveillances for the 1/4 inch vacuum tests do not include the restriction on plant conditions that requires the surveillances to be performed during a refueling outage, prior to refueling. These Surveillances could be adequately performed in other than a refueling outage without jeopardizing safe plant operations. The control of the plant conditions appropriate to perform the test is an issue for procedures and scheduling, and has been determined by the NRC Staff to

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.1 - SECONDARY CONTAINMENT

plant conditions appropriate to perform the test is an issue for procedures and scheduling, and has been determined by the NRC Staff to be unnecessary as a Technical Specification restriction. As indicated in Generic Letter 91-04, allowing this control is consistent with the vast majority of other Technical Specification surveillances that do not dictate plant conditions for the surveillances. The proposed change to the 18 month frequency also effectively increases the surveillance interval. The current Technical Specification for all three units requires performance at each refueling outage prior to refueling. Since the secondary containment is common to all three BFN units, with all three units operating, this could result in performance of the same test at an average of every 6 months. The change to the 18 month frequency will allow this test to be performed once and applied to all three units Technical Specifications. Since operating experience has shown these component usually pass the Surveillance at the 18 month frequency, the frequency is considered acceptable from a reliability standpoint.

- L2 Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operation and the inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown. By adding an exception to LCO 3.0.3 for the failing to suspend irradiated fuel movement, an LCO 3.0.3 required reactor shutdown is avoided in MODE 1, 2, or 3. However, the plant would still be required to shutdown after 4 hours per proposed Required Actions B.1 and B.2 in addition to suspending fuel movement per Required Action C.1.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



MAR 30 1990

3.7.B. Standby Gas Treatment System

4.7.B. Standby Gas Treatment System

3.7.B.4 (Cont'd)

- b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

See Justification for Changes for BFN 1STS 3.6.4.3

(A1)

~~3.7.C. Secondary Containment~~

4.7.C. Secondary Containment

(A2)
Proposed LCO 3.6.4.2 + Applicability

- 1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

- 1. Secondary containment surveillance shall be performed as indicated below:

(A3) Proposed Notes 2+3 to actions

Proposed Note 1 to Actions

(L1)

- a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system leakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

(L2)

- 2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:

Proposed Note to Required Action D.1

(ACTION D)

- a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel. *immediately* (M2)

- 2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

(L1)

(ACTIONS A+B)

- b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

(ACTION C)

See Justification for changes for BFN 1STS 3.6.4.1

(M1)

Proposed SRs 3.6.4.2.1, 3.6.4.2.2

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

MAR 30 1990

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.7.B. Standby Gas Treatment System

3.7.B.4 (Cont'd)

- b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

4.7.B. Standby Gas Treatment System

See Justification for Changes for BFN 1STS 3.6.4.3

3.7.C. Secondary Containment

(A1)

- 1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

4.7.C. Secondary Containment

- 1. Secondary containment surveillance shall be performed as indicated below:
 - a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system leakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.
- 2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

See Justification for Changes for BFN 1STS 3.6.4.1

Proposed SRS 3.6.4.2.1, 3.6.4.2.2

(M1)

(A2)

Proposed LCO 3.6.4.2 Applicability

(A3)

Proposed Notes 2+3 to ACTIONS

(L1)

Proposed Note 1 to ACTIONS

(L2)

Proposed Note to Required Action D.1

ACTION D

- 2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:
 - a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel. *immediately* (M1)
 - b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

(L1)

ACTIONS A+B

ACTION C

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

NOV 18 1991

A1

3.7.B. Standby Gas Treatment System

3.7.B.4 (Cont'd)

- b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

4.7.B. Standby Gas Treatment System

see justification for changes for BFN ISTS 3.6.4.3

3.7.C. Secondary Containment

Proposed LCO 3.6.4.2 + Applicability

- 1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

* LCO not applicable until just prior to loading fuel into the Unit 3 reactor vessel, provided the Unit 3 reactor zone is not required for secondary containment integrity for other units.

Proposed Notes 2 + 3 to Actions

Proposed Note 1 to Actions

- 2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:

Proposed Note to Required Action D.1

- a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel. *immediately*

- b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

4.7.C. Secondary Containment

- 1. Secondary containment surveillance shall be performed as indicated below:
 - a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system inleakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.
- 2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

see justification for changes for BFN ISTS 3.6.4.1

Proposed SRs 3.6.4.2.1, 3.6.4.2.2



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.2
SECONDARY CONTAINMENT ISOLATION VALVES

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The current definition of Secondary Containment Integrity requires all secondary containment isolation valves (SCIVs) to be OPERABLE or in their isolation position. Thus, the current secondary containment Specification encompasses the SCIV requirements. It is proposed to provide a separate Specification for SCIVs for clarity. Thus, the new LCO will require all SCIVs to be OPERABLE, consistent with the current requirements. The applicability has been reworded to be consistent with the new definitions of MODES and to have a positive statement as to when it is applicable, not when it is not applicable.

- A3 Proposed ACTIONS Note 2 ("Separate Condition entry is allowed for each penetration flow path") provides explicit instructions for proper application of the ACTIONS for Technical Specification compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," this Note provides direction consistent with the intent of the existing Actions for inoperable isolation valves. Similarly, proposed ACTIONS Note 3 facilitates the use and understanding of the intent to consider the operability of any system affected by inoperable isolation valves and to apply applicable Actions. With the proposed LCO 3.0.6, this intent would not necessarily apply. This clarification is consistent with the intent and interpretation of the existing Technical Specifications, and is therefore considered an administrative presentation preference.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.2
SECONDARY CONTAINMENT ISOLATION VALVES

- A4 Amendment 159 to Unit 3 Technical Specifications added a provision to allow separating the Unit 3 reactor zone from the secondary containment envelope under certain conditions (prior to fuel loading) to expedite Unit 3 construction activities during Unit 2 operation. This provision is no longer needed and can no longer be applied. Therefore the * Note to TS 3.7.C.1 has been deleted. This change is considered administrative since it deletes a requirement that no longer applies.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 Two new Surveillance Requirements have been added to ensure SCIV operability. SR 3.6.4.2.1 verifies that SCIVs isolate within the assumed times in accordance with the inservice testing program. SR 3.6.4.2.2 verifies that each SCIV actuates to its isolation position on an accident signal every 18 months. These are additional restrictions on plant operation.
- M2 This change requires the movement of irradiated fuel in secondary containment and CORE ALTERATIONS to be "immediately" suspended if secondary containment is inoperable. In addition, action must be "immediately" initiated to suspend operations with the potential to drain the reactor vessel in this Condition. The current specification does not establish a time limit to suspend these activities. Immediately suspending these activities minimizes the probability of a fission product release if a reactivity event occurs while the secondary containment is inoperable. Also, immediately initiating action to suspend operation with the potential to drain the reactor vessel will minimize the potential for reactor vessel draindown and subsequent potential for fission release. Imposing a time limit to suspended these activities is a more restrictive change.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Specific"

- L1 This Action has been changed to allow one valve in a penetration to be inoperable for up to 8 hours, instead of the current 4 hours. Proposed ACTION A now requires the penetration to be isolated in 8 hours. This is justified since an OPERABLE valve in the penetration remains to isolate the penetration if needed, thus the "leak tightness" of the secondary containment is still maintained. The isolated penetration is required to be verified every 31 days while a valve is inoperable, further ensuring the continued "leak tightness" of the secondary containment. Proposed ACTION B will verify that if both SCIVs in a penetration are inoperable, at least one SCIV in a penetration is closed

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.2
SECONDARY CONTAINMENT ISOLATION VALVES

within 4 hours. This maintains consistency with the current requirements. An allowance is proposed for intermittently opening closed secondary containment isolation valves under administrative control. The allowance is presented in proposed ACTIONS Note 1, which allows opening of secondary containment penetrations on an intermittent basis for performing Surveillances, repairs, routine evolutions, etc.

- L2 Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operation and the inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown. By adding an exception to LCO 3.0.3 for the failing to suspend irradiated fuel movement, an LCO 3.0.3 required reactor shutdown is avoided in MODE 1, 2, or 3. However, the plant would still be required to shutdown per proposed Required Actions C.1 and C.2 in addition to suspending fuel movement per Required Action D.1. However, this shutdown is considered less restrictive since Required Action C.1 allows the plant to be in Hot Shutdown within 12 hours versus Hot Standby within 6 hours as required by CTS 1.0.C.1. Both CTS and the proposed Required Action C.2 require the plant to be in Cold Shutdown within 36 hours.



UNIT 1

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



FEB 13 1995

~~3.7/4.7 CONTAINMENT SYSTEMS~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B. Standby Gas Treatment System~~

~~4.7.B. Standby Gas Treatment System~~

2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at $\geq 10\%$ design flow on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

b. The results of laboratory carbon sample analysis shall show $\geq 90\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.

c. System shall be shown to operate within $\pm 10\%$ design flow.

(A2)

2. a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once per operating cycle or once every 18 months whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.



MAR 30 1990

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B Standby Gas Treatment System~~

~~4.7.B Standby Gas Treatment System~~

(A4) Proposed ACTION D →

(A3) ON an actual or simulated initiation signal

ACTION A
3. From and after the date that one train of the standby gas treatment system is made or found to be inoperable for any reason, REACTOR POWER OPERATION and fuel handling is permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE, provided that during such 7 days all active components of the other two standby gas treatment trains shall be operable.

(L3) Proposed Note to Required Actions of C+E

(L2) Proposed Required Action C

ACTIONS C+E
4. If these conditions cannot be met:
a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel.

(M1) Immediately

~~4.7.B.2 (Cont'd)~~

SR 3.6.4.3.1

(M2) d. Each train shall be operated a total of at least 10 hours every month.

(LA1)

e. Test sealing of gaskets for housing doors shall be performed utilizing chemical smoke generators during each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a. (A5)

SR 3.6.4.3.2

3. a. Once per operating cycle, automatic initiation of each branch of the standby gas treatment system shall be demonstrated from each unit's controls. (18 months)

(LA2)

SR 3.6.4.3.4

b. At least once per year manual operability of the bypass valve for filter cooling shall be demonstrated. (12 months) (A1)

(L1)

c. When one train of the standby gas treatment system becomes inoperable the other two trains shall be demonstrated to be OPERABLE within 2 hours and daily thereafter.

(A1)

during movement of irradiated fuel assemblies in the secondary containment during CORE ALTERATION, or during OPRDRs

MAR 30 1990

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B. Standby Gas Treatment System~~ (A1)

~~4.7.B. Standby Gas Treatment System~~

~~3.7.B.4 (Cont'd)~~

ACTION
B

b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

see Justification for Change for BFN ISTS 3.6.4.1

3.7.C. Secondary Containment

4.7.C. Secondary Containment

- 1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

- 1. Secondary containment surveillance shall be performed as indicated below:

- a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system leakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

- 2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:
 - a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel.
 - b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

- 2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.



~~3.7/4.7 CONTAINMENT SYSTEMS~~

Specification 3.6.4.3
MAR 30 1990

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B. Standby Gas Treatment System~~

1. Except as specified in Specification 3.7.B.3 below, all three trains of the standby gas treatment system shall be OPERABLE at all times when secondary containment integrity is required.

LC 3.6.4.3

Applicability

See Justification for Change for BFN ISTS Section 5.0

~~4.7.B. Standby Gas Treatment System~~

1. At least once per year, the following conditions shall be demonstrated.
 - a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at a flow of 9000 cfm ($\pm 10\%$).
 - b. The inlet heaters on each circuit are tested in accordance with ANSI N510-1975, and are capable of an output of at least 40 kW.
 - c. Air distribution is uniform within 20% across HEPA filters and charcoal adsorbers.

A2 Proposed SR 3.6.4.3.2



(A1)

3.7.F. Primary Containment Purge System

4.7.F. Primary Containment Purge System

1. The primary containment purge system shall be OPERABLE for PURGING, except as specified in 3.7.F.2.
 - a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.
 - b. The results of laboratory carbon sample analysis shall show $\geq 85\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.
 - c. System flow rate shall be shown to be within $\pm 10\%$ of design flow when tested in accordance with ANSI N510-1975.
2. If the provisions of 3.7.F.1.a, b, and c cannot be met, the system shall be declared inoperable. The provisions of Technical Specification 1.C.1 do not apply. PURGING may continue using the Standby Gas Treatment System.

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8.5 inches of water at system design flow rate ($\pm 10\%$).
 - a. The tests and sample analysis of Specification 3.7.F.1 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first or after 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.
 - b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
 - c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

3. a. The 18-inch primary containment isolation valves associated with PURGING may be open during the RUN mode for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. The OPERABILITY of

See Justification for Changes for BFN 15TS 3.6.1.3

(R1)

UNIT 2

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B. Standby Gas Treatment System~~

(A1)

~~4.7.B. Standby Gas Treatment System~~

2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at $\geq 10\%$ design flow on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

b. The results of laboratory carbon sample analysis shall show $\geq 90\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.

c. System shall be shown to operate within $\pm 10\%$ design flow.

(A2)

2. a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once per operating cycle or once every 18 months whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

MAR 30 1990

~~3.7/A.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B. Standby Gas Treatment System~~ (A1)

~~4.7.B. Standby Gas Treatment System~~

~~4.7.B.2 (Cont'd)~~

~~SR 3.6.4.3.1~~

d. Each train shall be operated a total of at least 10 hours every month.

e. Test sealing of gaskets for housing doors shall be performed utilizing chemical smoke generators during each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a.

~~SR 3.6.4.3.3~~

3. a. ~~Once per operating cycle~~ ^{18 months} automatic initiation of each branch of the standby gas treatment system shall be demonstrated from each unit's controls.

~~SR 3.6.4.3.4~~

b. At least ^{12 months} ~~once per year~~ manual operability of the bypass valve for filter cooling shall be demonstrated.

c. When one train of the standby gas treatment system becomes inoperable the other two trains shall be demonstrated to be OPERABLE within 2 hours and daily thereafter.

during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPERATIONS

(A4) Proposed ACTION D →

(A3) on an actual or simulated initiation signal

ACTION A

3. From and after the date that one train of the standby gas treatment system is made or found to be inoperable for any reason, REACTOR POWER OPERATION and fuel handling is permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE, provided that during such 7 days all active components of the other two standby gas treatment trains shall be operable.

(L3) Proposed Notes to Required Actions of C+E:1

(L2) Proposed Required Action C:1 →

ACTIONS C+E

4. If these conditions cannot be met:
a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel immediately

(M1)

MAR 30 1990

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B. Standby Gas Treatment System~~

~~4.7.B. Standby Gas Treatment System~~

(A1)

~~3.7.B.4 (Cont'd)~~

ACTION B

- b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

See Justification for Changes for BFN ISTS 3.6.4.1

3.7.C. Secondary Containment

4.7.C. Secondary Containment

1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.
2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:
 - a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel.
 - b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

1. Secondary containment surveillance shall be performed as indicated below:
 - a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system leakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.
2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.



MAR 30 1990

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B. Standby Gas Treatment System~~

- 1. Except as specified in Specification 3.7.B.3 below, all three trains of the standby gas treatment system shall be OPERABLE at all times when secondary containment integrity is required.

LC0 36.4.3

Applicability

See Justification for Changes for BFN ISTS Section 5.0

~~4.7.B. Standby Gas Treatment System~~

- 1. At least once per year, the following conditions shall be demonstrated.
 - a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at a flow of 9000 cfm ($\pm 10\%$).
 - b. The inlet heaters on each circuit are tested in accordance with ANSI N510-1975, and are capable of an output of at least 40 kW.
 - c. Air distribution is uniform within 20% across HEPA filters and charcoal adsorbers.

(A2)

Proposal SR 3.6.4.3.2



(A1)

FEB 13 1995

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.F. Primary Containment Purge System

1. The primary containment purge system shall be OPERABLE for PURGING, except as specified in 3.7.F.2.
 - a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.
 - b. The results of laboratory carbon sample analysis shall show $\geq 85\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.
 - c. System flow rate shall be shown to be within $\pm 10\%$ of design flow when tested in accordance with ANSI N510-1975.
2. If the provisions of 3.7.F.1.a, b, and c cannot be met, the system shall be declared inoperable. The provisions of Technical Specification 1.C.1 do not apply. PURGING may continue using the Standby Gas Treatment System.

RI

T

3. a. The 18-inch primary containment isolation valves associated with PURGING may be open during the RUN mode for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. The OPERABILITY of

4.7.F. Primary Containment Purge System

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8.5 inches of water at system design flow rate ($\pm 10\%$).
 - a. The tests and sample analysis of Specification 3.7.F.1 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first or after 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.
 - b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
 - c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

See Justification for Changes for BFN 1STS 3.6.1.3



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



(A1)

~~3.7.B. Standby Gas Treatment System~~

1. Except as specified in Specification 3.7.B.3 below, all three trains of the standby gas treatment system shall be OPERABLE at all times when secondary containment integrity is required.

LCD 3.6.4.3

Applicability

See Justification for Changes for BFN ISTS Section 5.0

~~4.7.B. Standby Gas Treatment System~~

1. At least once per year, the following conditions shall be demonstrated.

- a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at a flow of 9000 cfm ($\pm 10\%$).
- b. The inlet heaters on each circuit are tested in accordance with ANSI N510-1975, and are capable of an output of at least 40 kW.
- c. Air distribution is uniform within 20% across HEPA filters and charcoal adsorbers.

(A2) Proposed SR 3.6.4.3.2



(A1)

~~3.7.B. Standby Gas Treatment System~~

~~4.7.B. Standby Gas Treatment System~~

2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at $\geq 10\%$ design flow on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

b. The results of laboratory carbon sample analysis shall show $\geq 90\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.

c. System shall be shown to operate within $\pm 10\%$ design flow.

2. a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once per operating cycle or once every 18 months whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

F

(A2)

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.B. Standby Gas Treatment System

4.7.B. Standby Gas Treatment System

Proposed Action D →

(A4)

(A3)

on an actual or simulated initiation signal

Action A

3. From and after the date that one train of the standby gas treatment system is made or found to be inoperable for any reason, REACTOR POWER OPERATION and fuel handling is permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE, provided that during such 7 days all active components of the other two standby gas treatment trains shall be operable.

(L3) Proposed Note to Required Actions of C+E.1

(L2) Proposed Required Action C.1 →

4. If these conditions cannot be met:

Actions C+E

a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel

immediately (M1)

4.7.B.2 (Cont'd)

SR 3.6.4.3.1

(M2)

d. Each train shall be operated a total of at least 10 hours every month.

e. Test sealing of gaskets for housing doors shall be performed utilizing chemical smoke generators during each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a.

(LA1)

SR 3.6.4.3.2

(A5) 12 months

3. a. Once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated from each unit's controls

(LA2)

SR 3.6.4.3.4

(A1) 12 months

b. At least once per year manual operability of the bypass valve for filter cooling shall be demonstrated.

c. When one train of the standby gas treatment system becomes inoperable the other two trains shall be demonstrated to be OPERABLE within 2 hours and daily thereafter.

(L1)

(A1)

during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPRVs



(A1)

~~3.7.B. Standby Gas Treatment System~~

~~4.7.B. Standby Gas Treatment System~~

~~3.7.B.4 (Cont'd)~~

Action B

- b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

See Justification for Changes for BFN ISTS 3.6.4.1

3.7.C. Secondary Containment

4.7.C. Secondary Containment

- * 1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.
- * LCO not applicable until just prior to loading fuel into the Unit 3 reactor vessel, provided the Unit 3 reactor zone is not required for secondary containment integrity for other units.
- 2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:
 - a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel.
 - b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

- 1. Secondary containment surveillance shall be performed as indicated below:
 - a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system inleakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.
- 2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.



(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.7.F. Primary Containment Purge System

4.7.F. Primary Containment Purge System

1. The primary containment purge system shall be OPERABLE for PURGING, except as specified in 3.7.F.2.
 - a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.
 - b. The results of laboratory carbon sample analysis shall show $\geq 85\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.
 - c. System flow rate shall be shown to be within $\pm 10\%$ of design flow when tested in accordance with ANSI N510-1975.
2. If the provisions of 3.7.F.1.a, b, and c cannot be met, the system shall be declared inoperable. The provisions of Technical Specification 1.C.1 do not apply. PURGING may continue using the Standby Gas Treatment System.

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8.5 inches of water at system design flow rate ($\pm 10\%$).
 - a. The tests and sample analysis of Specification 3.7.F.1 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first or after 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.
 - b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
 - c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

3. a. The 18-inch primary containment isolation valves associated with PURGING may be open during the RUN mode for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. The OPERABILITY of

See justification for changes for BFN 15TS 3.6.1.3

(R1) →

+



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.3 - STANDBY GAS TREATMENT SYSTEM

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The technical content of this requirement is being moved to Section 5.0 of the proposed Technical Specifications in accordance with the format of the BWR Standard Technical Specifications, NUREG 1433. Any technical changes to this requirement will be addressed within the content of proposed Specification 5.5.7. A surveillance requirement (proposed SR 3.6.4.3.2) is added to clarify that the tests of the Ventilation Filter Testing Program must also be completed and passed for determining operability of the SGT System. Since this is a presentation preference that maintains current requirements, this change is considered administrative.
- A3 The description of the signal used to automatically initiate the SGT System "actual or simulated initiation signal" has been added for clarity. This is consistent with the BWR Standard Technical Specifications, NUREG 1433, and no change is intended.
- A4 A new ACTION is proposed (ACTION D) which directs entry into LCO 3.0.3 if two or more required standby gas treatment subsystems are inoperable in Modes 1, 2, or 3. This avoids confusion as to the proper action if in Modes 1, 2, or 3 and simultaneously handling fuel, conducting CORE ALTERATIONS, or operations with the potential for draining the reactor vessel. Since the proposed ACTION effectively results in the same action as the current specification, this change is considered administrative.
- A5 The Frequency for verifying SGTS automatic initiation has been changed to 18 months from once per operating cycle. The BFN operating cycle is currently defined as 18 months. As such this is a change in presentation only and is therefore administrative.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.3 - STANDBY GAS TREATMENT SYSTEM

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 This change requires the movement of irradiated fuel in secondary containment and CORE ALTERATIONS to be "Immediately" suspended if secondary containment is inoperable. In addition, action must be "Immediately" initiated to suspend operations with the potential to drain the reactor vessel in this Condition. The current specification does not establish a time limit to suspend these activities. Immediately suspending these activities minimizes the probability of a fission product release if a reactivity event occurs while the secondary containment is inoperable. Also, immediately initiating action to suspend operation with the potential to drain the reactor vessel will minimize the potential for reactor vessel draindown and subsequent potential for fission release. Imposing a time limit to suspended these activities is a more restrictive change.
- M2 CTS 4.7.B.2.d requires each train to be operated a total of at least 10 hours each month. Proposed SR 3.6.4.3.1 requires each train to be operated continuously for 10 hours. As such, the proposed SR is considered more restrictive.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 Details on methods of testing gasket seals for housing doors has been deleted. This type of detail will be retained in plant procedures and/or system operating instructions.
- LA2 Details on the method of performing Standby Gas Treatment system surveillance requirements have been relocated to plant procedures. Changes to the procedure will be controlled by licensee controlled programs.

"Specific"

- L1 The proposed change will delete the requirement to test the other SGT subsystems when one subsystem is inoperable. The requirement for demonstrating operability of the redundant subsystems was originally chosen because there was a lack of plant operating history and a lack of sufficient equipment failure data. Since that time, plant operating experience has demonstrated that testing of the redundant subsystems when one subsystem is inoperable is not necessary to provide adequate assurance of system operability.

This change will allow credit to be taken for normal periodic surveillances as a demonstration of operability and availability of the

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.3 - STANDBY GAS TREATMENT SYSTEM

remaining components. The periodic frequencies specified to demonstrate operability of the remaining components have been shown to be adequate to ensure equipment operability. As stated in NRC Generic Letter 87-09, "It is overly conservative to assume that systems or components are inoperable when a surveillance requirement has not been performed. The opposite is in fact the case; the vast majority of surveillances demonstrate the systems or components in fact are operable." Therefore, reliance on the specified surveillance intervals does not result in a reduced level of confidence concerning the equipment availability. Also, the current Standard Technical Specifications (STS) and, more specifically, all the Technical Specifications approved for recently licensed BWRs accept the philosophy of system operability based on satisfactory performance of monthly, quarterly, refueling interval, post maintenance or other specified performance tests without requiring additional testing when another system is inoperable (except for diesel generator testing, which is not being changed).

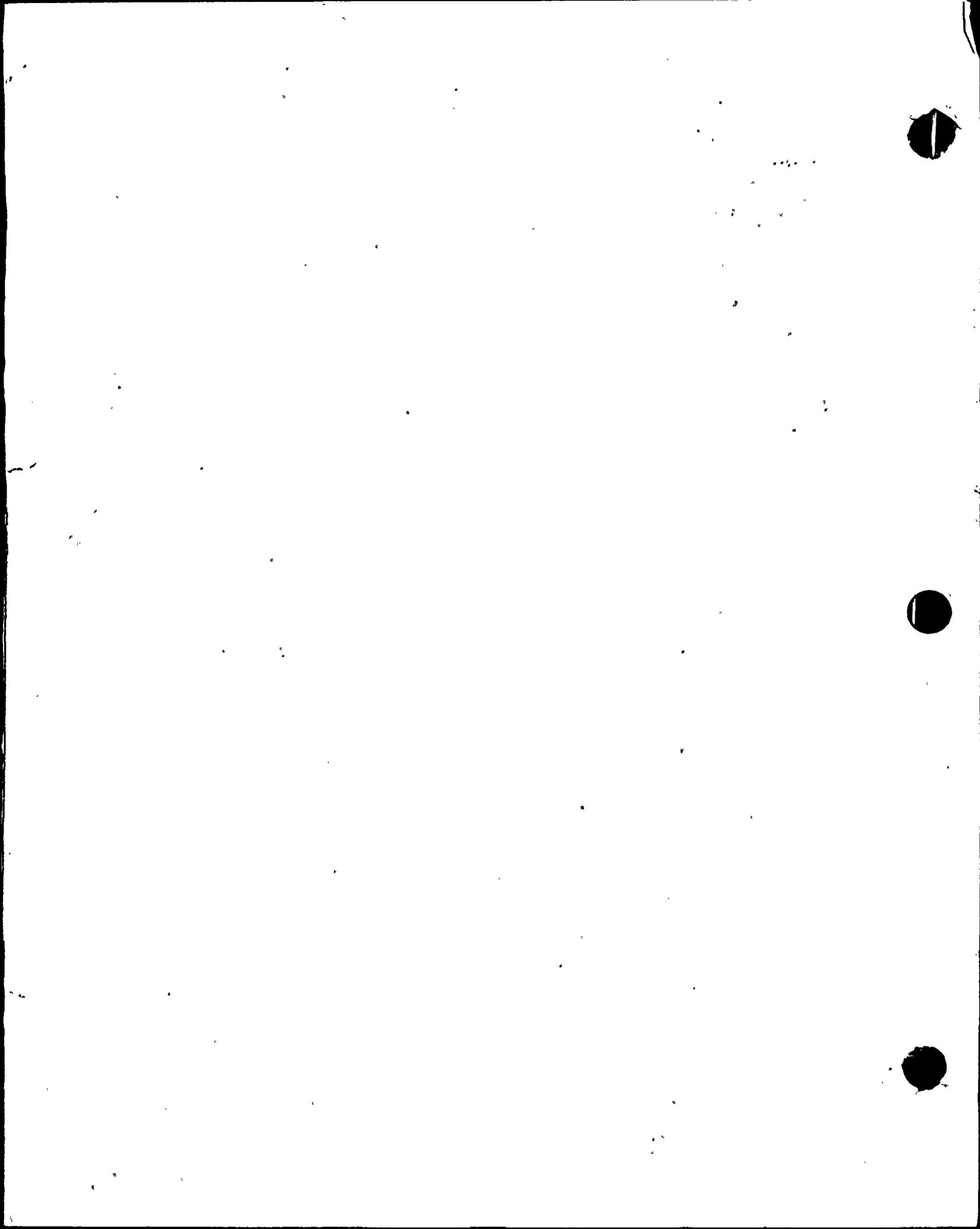
- L2 An alternative is proposed to suspending operations if a SGT subsystem cannot be returned to OPERABLE status within seven days, and movement of irradiated fuel assemblies, CORE ALTERATIONS, or operations with the potential for draining the reactor vessel are being conducted. The alternative is to initiate two OPERABLE subsystems of SGT and continue to conduct the operations. Since two subsystems are sufficient for any accident, the risk of failure of the subsystems to perform their intended function is significantly reduced if they are running. This alternative is less restrictive than the existing requirement. However, the proposed alternative ensures that the remaining subsystems are Operable, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected. This change is consistent with NUREG-1433.
- L3 The Required Actions of C and E.1 have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operation and the inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown. By adding an exception to LCO 3.0.3 for the failing to suspend irradiated fuel movement, an LCO 3.0.3 required reactor shutdown is avoided in MODE 1, 2, or 3. However, the plant would still be required to shutdown per proposed Required Actions B.1 and B.2 in addition to suspending fuel movement per Required Actions C.1 and E.1. However, this shutdown is considered less restrictive since Required Action B.1 allows the plant to be in Hot Shutdown within 12 hours versus Hot Standby within 6 hours as required by CTS 1.0.C.1. Both CTS and the proposed Required Action B.2 require the plant to be in Cold Shutdown within 36 hours.



JUSTIFICATION FOR CHANGES
CTS 3.7.F/4.7.F - PRIMARY CONTAINMENT PURGE SYSTEM

RELOCATED CHANGES

- R1 CTS 3.7.F.1 & 2 and 4.7.F requirements have been relocated to the Technical Requirements Manual (TRM). The Primary Containment Purge System is normally isolated and normally not required to be functional during power operation. It does provide the preferred exhaust path for purging the primary containment; however, the SGTS can be used to perform the equivalent function. The supply and isolation valves are depended on to function properly for containment isolation, which is covered in proposed BFN ISTS Section 3.6.1.3, Primary Containment Isolation Valves.



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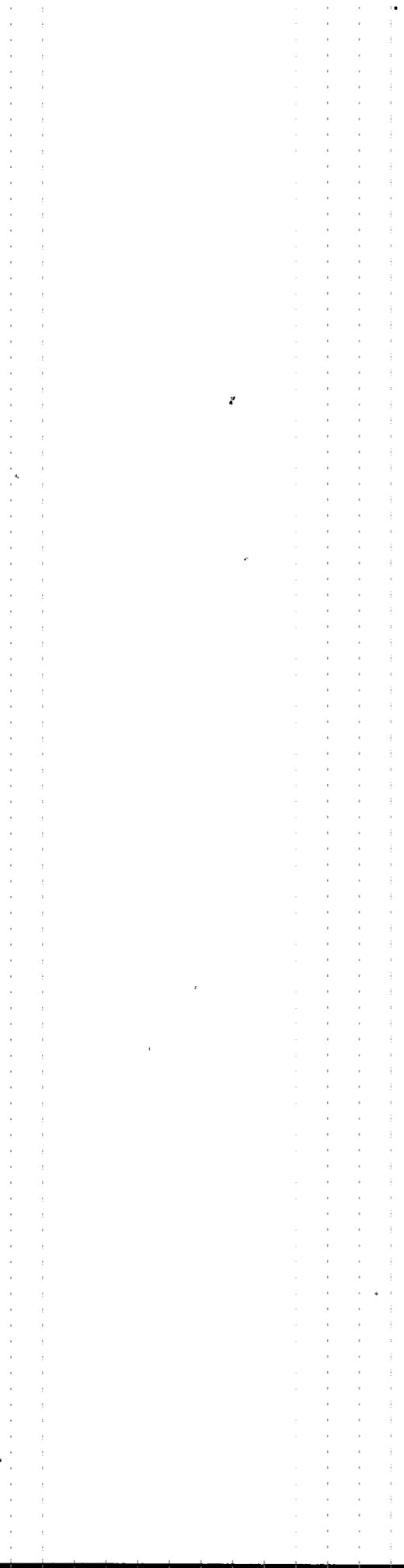
BROWNS FERRY NUCLEAR PLANT

IMPROVED STANDARD TECHNICAL SPECIFICATIONS

Enclosure III
Volume 7

TENNESSEE VALLEY AUTHORITY

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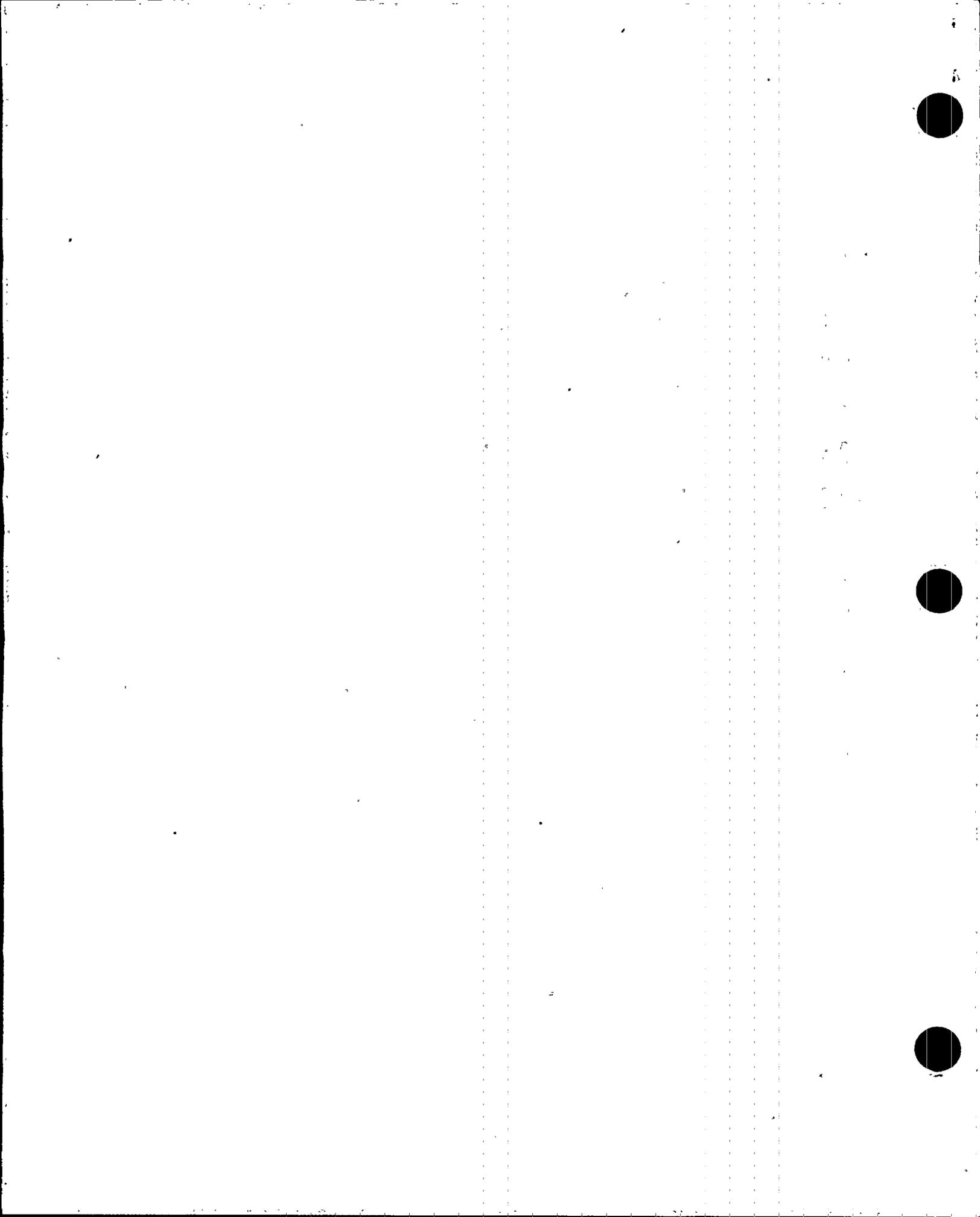
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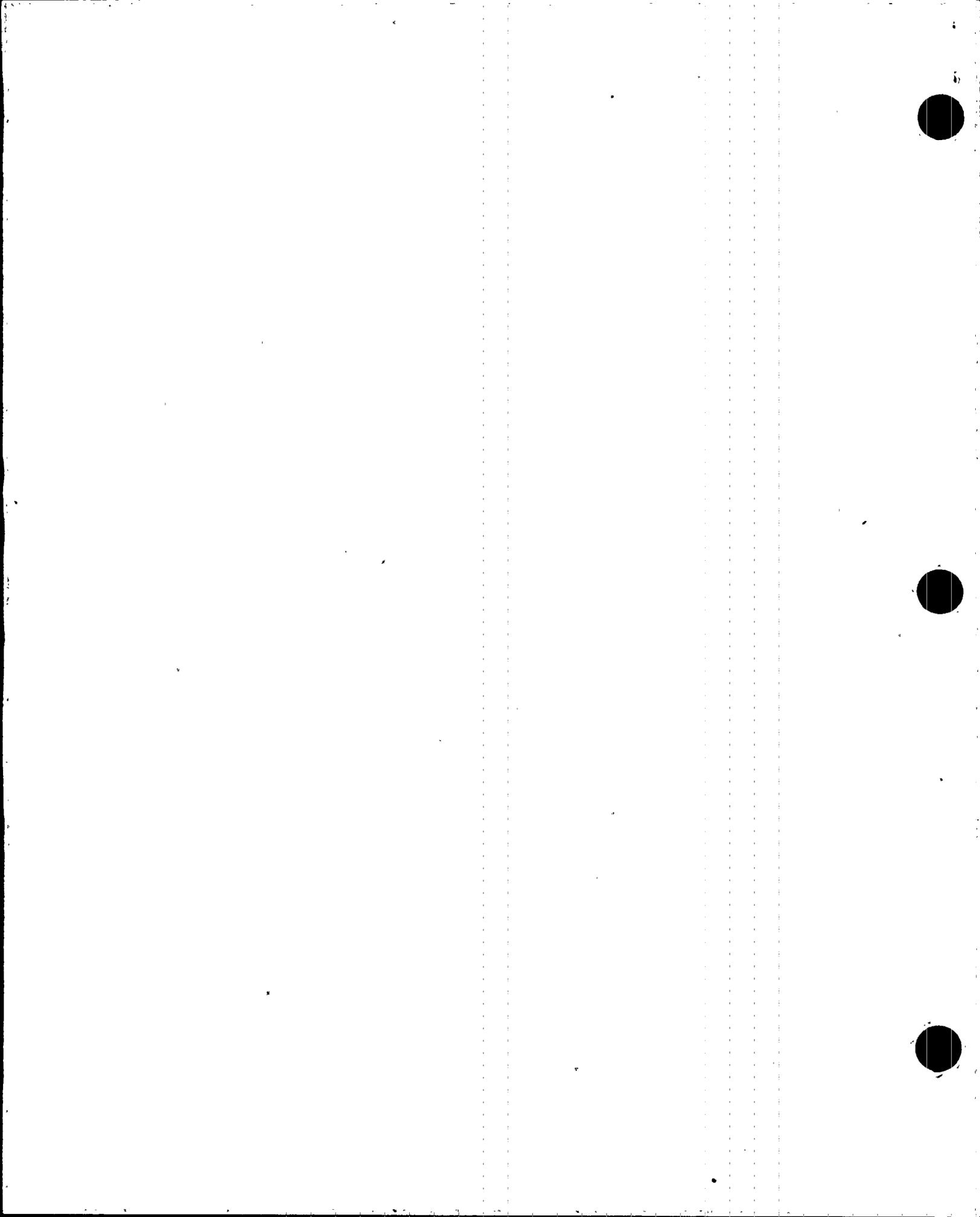
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TABLE OF CONTENTS

<u>Section</u>	<u>Page No.</u>
B 2.0 SAFETY LIMITS (SLs)	B 2.0-1
B 2.1.1 Reactor Core SLs	B 2.0-1
B 2.1.2 Reactor Coolant System (RCS) Pressure SL	B 2.0-7
B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY	B 3.0-1
B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY	B 3.0-10
B 3.1 REACTIVITY CONTROL SYSTEMS	B 3.1-1
B 3.1.1 SHUTDOWN MARGIN (SDM)	B 3.1-1
B 3.1.2 Reactivity Anomalies	B 3.1-8
B 3.1.3 Control Rod OPERABILITY	B 3.1-13
B 3.1.4 Control Rod Scram Times	B 3.1-22
B 3.1.5 Control Rod Scram Accumulators	B 3.1-29
B 3.1.6 Rod Pattern Control	B 3.1-34
B 3.1.7 Standby Liquid Control (SLC) System	B 3.1-59
B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves	B 3.1-46
B 3.2 POWER DISTRIBUTION LIMITS	B 3.2-1
B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	B 3.2-1
B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)	B 3.2-4
B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)	B 3.2-9
B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints	B 3.2-12
B 3.3 INSTRUMENTATION	B 3.3-1
B 3.3.1.1 Reactor Protection System (RPS) Instrumentation	B 3.3-1
B 3.3.1.2 Source Range Monitor (SRM) Instrumentation	B 3.3-33
B 3.3.2.1 Control Rod Block Instrumentation	B 3.3-42
B 3.3.2.2 Feedwater and Main Turbine Trip Instrumentation	B 3.3-53
B 3.3.3.1 Post Accident Monitoring (PAM) Instrumentation	B 3.3-60
B 3.3.3.2 Backup Control System	B 3.3-71
B 3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation	B 3.3-80
B 3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation	B 3.3-89
B 3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation	B 3.3-98
B 3.3.5.2 Reactor Core Isolation Cooling (RCIC) System Instrumentation	B 3.3-135
B 3.3.6.1 Primary Containment Isolation Instrumentation	B 3.3-143
B 3.3.6.2 Secondary Containment Isolation Instrumentation	B 3.3-165
B 3.3.7.1 Control Room Emergency Ventilation (CREV) System Instrumentation	B 3.3-176
B 3.3.8.1 Loss of Power (LOP) Instrumentation	B 3.3-188
B 3.3.8.2 Reactor Protection System (RPS) Electric Power Monitoring	B 3.3-196



<u>Section</u>		<u>Page No.</u>
B 3.4	REACTOR COOLANT SYSTEM (RCS)	B 3.4-1
B 3.4.1	Recirculation Loops Operating	B 3.4-1
B 3.4.2	Jet Pumps	B 3.4-9
B 3.4.3	Safety/Relief Valves (S/RVs)	B 3.4-14
B 3.4.4	RCS Operational LEAKAGE	B 3.4-19
B 3.4.5	RCS Leakage Detection Instrumentation	B 3.4-25
B 3.4.6	RCS Specific Activity	B 3.4-31
B 3.4.7	Residual Heat Removal (RHR) Shutdown Cooling System-Hot Shutdown	B 3.4-35
B 3.4.8	Residual Heat Removal (RHR) Shutdown Cooling System-Cold Shutdown	B 3.4-40
B 3.4.9	RCS Pressure and Temperature (P/T) Limits	B 3.4-45
B 3.5	EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM	B 3.5-1
B 3.5.1	ECCS-Operating	B 3.5-1
B 3.5.2	ECCS-Shutdown	B 3.5-18
B 3.5.3	RCIC System	B 3.5-24
B 3.6	CONTAINMENT SYSTEMS	B 3.6-1
B 3.6.1.1	Primary Containment	B 3.6-1
B 3.6.1.2	Primary Containment Air Lock	B 3.6-6
B 3.6.1.3	Primary Containment Isolation Valves (PCIVs)	B 3.6-14
B 3.6.1.4	Drywell Air Temperature	B 3.6-28
B 3.6.1.5	Reactor Building-to-Suppression Chamber Vacuum Breakers	B 3.6-31
B 3.6.1.6	Suppression Chamber-to-Drywell Vacuum Breakers	B 3.6-37
B 3.6.2.1	Suppression Pool Average Temperature	B 3.6-43
B 3.6.2.2	Suppression Pool Water Level	B 3.6-49
B 3.6.2.3	Residual Heat Removal (RHR) Suppression Pool Cooling	B 3.6-52
B 3.6.2.4	Residual Heat Removal (RHR) Suppression Pool Spray	B 3.6-57
B 3.6.2.5	Residual Heat Removal (RHR) Drywell Spray	B 3.6-75
B 3.6.2.6	Drywell-to-Suppression Chamber Differential Pressure	B 3.6-67
B 3.6.3.1	Containment Atmosphere Dilution (CAD) System	B 3.6-70
B 3.6.3.2	Primary Containment Oxygen Concentration	B 3.6-75
B 3.6.4.1	Secondary Containment	B 3.6-78
B 3.6.4.2	Secondary Containment Isolation Valves (SCIVs)	B 3.6-83
B 3.6.4.3	Standby Gas Treatment (SGT) System	B 3.6-89
B 3.7	PLANT SYSTEMS	B 3.7-1
B 3.7.1	Residual Heat Removal Service Water (RHRSW) System	B 3.7-1
B 3.7.2	Emergency Equipment Cooling Water (EECW) System and Ultimate Heat Sink (UHS)	B 3.7-7
B 3.7.3	Control Room Emergency Ventilation (CREV) System	B 3.7-12
B 3.7.4	Control Room Air Conditioning (AC) System	B 3.7-19
B 3.7.6	Main Condenser Offgas	B 3.7-30
B 3.7.7	Main Turbine Bypass System	B 3.7-24
B 3.7.8	Spent Fuel Storage Pool Water Level	B 3.7-28



<u>Section</u>		<u>Page No.</u>
B 3.8	ELECTRICAL POWER SYSTEMS	B 3.8-1
B 3.8.1	AC Sources - Operating	B 3.8-1
B 3.8.2	AC Sources - Shutdown	B 3.8-28
B 3.8.3	Diesel Fuel Oil, Lube Oil, and Starting Air	B 3.8-35
B 3.8.4	DC Sources - Operating	B 3.8-42
B 3.8.5	DC Sources - Shutdown	B 3.8-51
B 3.8.6	Battery Cell Parameters	B 3.8-55
B 3.8.7	Inverters - Operating	B 3.8-62
B 3.8.8	Inverters - Shutdown	B 3.8-73
B 3.9	REFUELING OPERATIONS	B 3.9-1
B 3.9.1	Refueling Equipment Interlocks	B 3.9-1
B 3.9.2	Refuel Position One-Rod-Out Interlock	B 3.9-5
B 3.9.3	Control Rod Position	B 3.9-9
B 3.9.4	Control Rod Position Indication	B 3.9-12
B 3.9.5	Control Rod OPERABILITY - Refueling	B 3.9-16
B 3.9.6	Reactor Pressure Vessel (RPV) Water Level	B 3.9-19
B 3.9.7	Residual Heat Removal (RHR) - High Water Level	B 3.9-22
B 3.9.8	Residual Heat Removal (RHR) - Low Water Level	B 3.9-26
B 3.10	SPECIAL OPERATIONS	B 3.10-1
B 3.10.1	Inservice Leak and Hydrostatic Testing Operation	B 3.10-1
B 3.10.2	Reactor Mode Switch Interlock Testing	B 3.10-6
B 3.10.3	Single Control Rod Withdrawal - Hot Shutdown	B 3.10-11
B 3.10.4	Single Control Rod Withdrawal - Cold Shutdown	B 3.10-16
B 3.10.5	Single Control Rod Drive (CRD) Removal - Refueling	B 3.10-21
B 3.10.6	Multiple Control Rod Withdrawal - Refueling	B 3.10-26
B 3.10.7	Control Rod Testing - Operating	B 3.10-29
B 3.10.8	SHUTDOWN MARGIN (SDM) Test - Refueling	B 3.10-33



B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

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

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

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

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

(continued)



BASES

BACKGROUND
(continued)

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

APPLICABLE
SAFETY ANALYSES

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

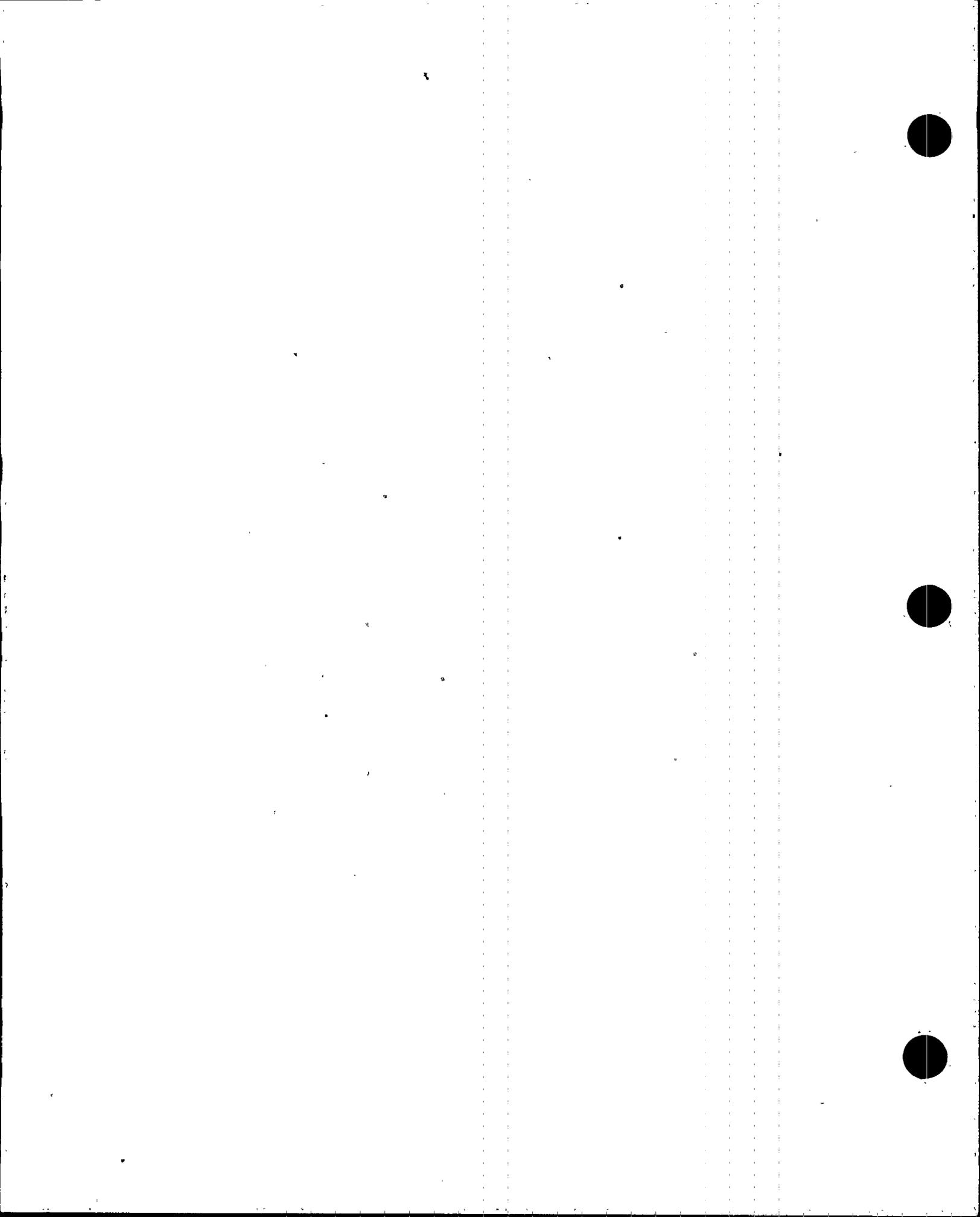
The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result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

2.1.1.1 Fuel Cladding Integrity

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

The static head across the fuel bundles due only to elevation effects from liquid only in the channel, core bypass region, and annulus at zero power, zero flow is approximately 4.5 psi. At all operating conditions, this pressure differential is maintained by the bypass region of the core and the annulus region of the vessel. The elevation head provided by the annulus produces natural circulation flow conditions which have balancing pressure head and loss terms

(continued)



BASES

inside the core shroud. This natural circulation principle maintains a core plenum to plenum pressure drop of about 4.5 to 5 psid along the natural circulation flow line of the P/F operating map. In the range of power levels of interest, approaching 25% of rated power below which thermal margin monitoring is not required, the pressure drop and density head terms tradeoff for power changes such that natural circulation flow is nearly independent of reactor power. This characteristic is represented by the nearly vertical portion of the natural circulation line on the P/F operating map. Analysis has shown that the hot channel flow rate is >28,000 lb/hr in the region of operation with power ~25% and core pressure drop of about 4.5 to 5 psid. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at 28,000 lb/hr is approximately 3 MWt. With the design peaking factors, this corresponds to a core thermal power of more than 50%. Thus operation up to 25% of rated power with normal natural circulation available is conservatively acceptable even if reactor pressure is equal to or below 800 psia (the limit of the range of applicability of GETAB/GEXL for GE fuel). If reactor power is significantly less than 25% of rated (e.g., below 10% of rated), the core flow and the channel flow supported by the available driving head may be less than 28,000 lb/hr (along the lower portion of the natural circulation flow characteristic on the P/F map). However, the critical power that can be supported by the core and hot channel flow with normal natural circulation paths available remains well above the actual power conditions. The inherent characteristics of BWR natural circulation make power and core flow follow the natural circulation line as long as normal water level is maintained.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

Thus, operation with core thermal power below 25% of rated without thermal margin surveillance is conservatively acceptable even for reactor operations at natural circulation. Adequate fuel thermal margins are also maintained without further surveillance for the low power conditions that would be present if core natural circulation is below 10% of rated flow (the limit of applicability of the GETAB/GEXL correlations for GE fuel).

2.1.1.2 MCPR

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

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

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

2.1.1.3 Reactor Vessel Water Level

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

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2 and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT
VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 3). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

(continued)



BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
 2. GE SIL No. 516, Supplement 2, January 19, 1996.
 3. 10 CFR 100.
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B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

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

During normal operation and abnormal operational 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 are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

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

(continued)



BASES (continued)

APPLICABLE
SAFETY ANALYSES

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

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME, Boiler and Pressure Vessel Code, 1965 Edition, including Addenda through the summer of 1965 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to the USAS Nuclear Power Piping Code, Section B31.1, 1967 Edition (Ref. 6), and the additional requirements of GE design and procurement specifications (Ref. 7) which were implemented in lieu of the outdated B31 Nuclear Code Cases - N2, N7, N9, and N10, for the reactor recirculation piping, which permits a maximum pressure transient of 120% of design pressures of 1148 psig for suction piping and 1326 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 120% of design pressures of 1148 psig for suction piping and 1326 psig for discharge piping. When the 20% allowance (230 and 265 psig) allowed by USAS Piping Code, Section B31.1, for pressure transients is added to the design pressures, transient pressure limits of 1378 and 1591 psig are established. The most limiting of these allowances is the 110% of design pressure for the RCS pressure vessel; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

(continued)



BASES (continued)

SAFETY LIMIT
VIOLATIONS

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
 2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
 3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
 4. 10 CFR 100.
 5. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, Summer of 1965 Addenda.
 6. ASME, USAS, Nuclear Power Piping Code, Section B31.1, 1967 Edition.
 7. BFN General Electric Design Specification 22A1406, "Pressure Integrity of Piping and Equipment Pressure Parts," Revision 2, April 28, 1970.
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B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs LCO 3.0.1 through LCO 3.0.7 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 3.4.9, "RCS Pressure and Temperature 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/divisions 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

(continued)



BASES

LCO 3.0.2 (continued) apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.

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

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

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BASES

LCO 3.0.3
(continued)

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

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

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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 3.7.6, "Spent Fuel Storage Pool Water Level." LCO 3.7.6 has an Applicability of "During movement of irradiated fuel assemblies in the spent fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.6 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 3.7.6 of "Suspend movement of irradiated fuel assemblies in the spent fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

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 different 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 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

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BASES

LCO 3.0.4
(continued)

practice of restoring systems or components to OPERABLE status before unit startup.

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 3 from MODE 4, MODE 2 from MODE 3 or 4, 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 MODE 1, 2, or 3. The requirements of LCO 3.0.4 do not apply in MODES 4 and 5, or in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual specifications sufficiently define the remedial measures to be taken. [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.]

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, either 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.

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BASES (continued)

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

- a. The OPERABILITY of the equipment being returned to service; or
- b. The OPERABILITY of other equipment; or
- c. That variables are within limits.

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 testing. 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 when a supported system LCO is not met solely due to a support system LCO not being met. This exception is provided because LCO 3.0.2 would require that the Conditions and

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BASES

LCO 3.0.6
(continued)

Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the plant is maintained in a safe condition are specified in the support systems' LCO's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCO's Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

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

Specification 5.5.11, "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.

The SFDP requires cross division checks to identify a loss of safety function for those support systems that support safety systems are required. The cross division check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. 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.

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BASES (continued)

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Special Operations LCOs in Section 3.10 allow specified TS requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Special Operations LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Special Operations LCOs is optional. A special operation may be performed either under the provisions of the appropriate Special Operations LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Special Operations LCO, the requirements of the Special Operations LCO shall be followed. When a Special Operations LCO requires another LCO to be met, only the requirements of the LCO statement are required to be met regardless of that LCO's Applicability (i.e., should the requirements of this other LCO not be met, the ACTIONS of the Special Operations LCO apply, not the ACTIONS of the other LCO). However, there are instances where the Special Operations LCO's ACTIONS may direct the other LCO's ACTIONS be met. The Surveillances of the other LCO are not required to be met, unless specified in the Special Operations LCO. If conditions exist such that the Applicability of any other LCO is met, all the other LCO's requirements (ACTIONS and SRs) are required to be met concurrent with the requirements of the Special Operations LCO.



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 to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a Special Operations LCO are only applicable when the Special Operations LCO is used as an allowable exception to the requirements of a Specification.

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.

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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.

Some examples of this process are:

- a. Control Rod Drive maintenance during refueling that requires scram testing at > 800 psi. However, if other appropriate testing is satisfactorily completed and the scram time testing of SR 3.1.4.3 is satisfied, the control rod can be considered OPERABLE. This allows startup to proceed to reach 800 psi to perform other necessary testing.
- b. High pressure coolant injection (HPCI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPCI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g.,

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BASES

SR 3.0.2
(continued)

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. 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

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BASES

SR 3.0.3
(continued)

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 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

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BASES

SR 3.0.3
(continued)

outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

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

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

(continued)



BASES

SR 3.0.4
(continued)

that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

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, 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 3 from MODE 4, MODE 2 from MODE 3 or 4, 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 MODE 1, 2, or 3. The requirements of SR 3.0.4 do not apply in MODES 4 and 5, or in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.



B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure:

- a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Events;
- b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and
- c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

These requirements are satisfied by the control rods, as described in GDC 26 (Ref. 1), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions.

APPLICABLE SAFETY ANALYSES

The control rod drop accident (CRDA) analysis (Refs. 2 and 3) assumes the core is subcritical with the highest worth control rod withdrawn. Typically, the first control rod withdrawn has a very high reactivity worth and, should the core be critical during the withdrawal of the first control rod, the consequences of a CRDA could exceed the fuel damage limits for a CRDA (see Bases for LCO 3.1.6, "Rod Pattern Control"). Also, SDM is assumed as an initial condition for the control rod removal error during refueling (Ref. 4) and fuel assembly insertion error during refueling (Ref. 5) accidents. The analysis of these reactivity insertion events assumes the refueling interlocks are OPERABLE when the reactor is in the refueling mode of operation. These interlocks prevent the withdrawal of more than one control rod from the core during refueling. (Special consideration and requirements for multiple control rod withdrawal during refueling are covered in Special Operations LCO 3.10.6, "Multiple Control Rod Withdrawal - Refueling.") The analysis assumes this condition is acceptable since the core will be

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

shut down with the highest worth control rod withdrawn, if adequate SDM has been demonstrated.

Prevention or mitigation of reactivity insertion events is necessary to limit energy deposition in the fuel to prevent significant fuel damage, which could result in undue release of radioactivity. Adequate SDM ensures inadvertent criticalities and potential CRDAs involving high worth control rods (namely the first control rod withdrawn) will not cause significant fuel damage.

SDM satisfies Criterion 2 of the NRC Policy Statement (Ref. 8).

LCO

The specified SDM limit accounts for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically or by measurement. This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement. When SDM is demonstrated by calculations not associated with a test (e.g., to confirm SDM during the fuel loading sequence), additional margin is included to account for uncertainties in the calculation. To ensure adequate SDM during the design process, a design margin is included to account for uncertainties in the design calculations (Ref. 6).

APPLICABILITY

In MODES 1 and 2, SDM must be provided because subcriticality with the highest worth control rod withdrawn is assumed in the CRDA analysis (Ref. 2). In MODES 3 and 4, SDM is required to ensure the reactor will be held subcritical with margin for a single withdrawn control rod. SDM is required in MODE 5 to prevent an open vessel, inadvertent criticality during the withdrawal of a single control rod from a core cell containing one or more fuel assemblies or a fuel assembly insertion error (Ref. 5).

(continued)



BASES (continued)

ACTIONS

A.1

With SDM not within the limits of the LCO in MODE 1 or 2, SDM must be restored within 6 hours. Failure to meet the specified SDM may be caused by a control rod that cannot be inserted. The allowed Completion Time of 6 hours is acceptable, considering that the reactor can still be shut down, assuming no failures of additional control rods to insert, and the low probability of an event occurring during this interval.

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 in 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1

With SDM not within limits in MODE 3, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core.

D.1, D.2, D.3, and D.4

With SDM not within limits in MODE 4, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least two Standby Gas Treatment (SGT) subsystems are OPERABLE; and secondary containment isolation capability (i.e., at least one secondary containment isolation valve (damper) and

(continued)



BASES

ACTIONS

D.1, D.2, D.3, and D.4 (continued)

associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability) in each associated secondary containment penetration flow path with isolation valve(s) (damper(s)) not isolated that is assumed to be isolated to mitigate radioactive releases. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

E.1, E.2, E.3, E.4, and E.5

With SDM not within limits in MODE 5, the operator must immediately suspend CORE ALTERATIONS that could reduce SDM (e.g., insertion of fuel in the core or the withdrawal of control rods). Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and are therefore excluded from the suspended actions.

Action must also be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and therefore do not have to be inserted.

Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least two SGT subsystems are OPERABLE; and secondary containment isolation capability (i.e., at least one secondary containment isolation valve (damper) and associated instrumentation are OPERABLE, or other acceptable

(continued)



BASES

ACTIONS

E.1, E.2, E.3, E.4, and E.5 (continued)

administrative controls to assure isolation capability) in each associated secondary containment penetration flow path with isolation valve(s) (damper(s)) not isolated that is assumed to be isolated to mitigate radioactivity releases. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the SRs needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Action must continue until all required components are OPERABLE.

SURVEILLANCE
REQUIREMENTSSR 3.1.1.1

Adequate SDM must be verified to ensure that the reactor can be made subcritical from any initial operating condition. This can be accomplished by a test, an evaluation, or a combination of the two. Adequate SDM is demonstrated before or during the first startup after fuel movement, or shuffling within the reactor pressure vessel, or control rod replacement. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 7). For the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10% $\Delta k/k$) must be added to the SDM limit of 0.28% $\Delta k/k$ to account for uncertainties in the calculation.

(continued)



BASES

SURVEILLANCE
REQUIREMENTSSR 3.1.1.1 (continued)

The SDM may be demonstrated during an in sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local criticals, where the highest worth control rod is determined by testing. Local critical tests require the withdrawal of out of sequence control rods. This testing would therefore require bypassing of the rod worth minimizer to allow the out of sequence withdrawal, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing - Operating").

The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. A spiral reload sequence does not preclude the practice of bridging between SRMs and filling in the center in order to provide for conservative core monitoring during core alterations. Removing fuel from the core will always result in an increase in SDM.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section 14.6.2.

(continued)



BASES

REFERENCES
(continued)

3. NEDE-24011-P-A-11-US, "General Electric Standard Application for Reactor Fuel," Supplement for United States, Section S.2.2.3.1, November 1995.
 4. FSAR, Section 14.5.3.3.
 5. FSAR, Section 14.5.3.4.
 6. FSAR, Section 3.6.5.2.
 7. NEDE-24011-P-A-11, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, November 1995.
 8. NRC 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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(continued)



B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

In accordance with GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and abnormal operational transients. Therefore, reactivity anomaly is used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity or control rod worth or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in assuring the reactor can be brought safely to cold, subcritical conditions.

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.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel loaded in the previous cycles provide excess positive reactivity beyond that required to sustain steady state operation at the beginning of cycle (BOC). When the reactor is critical at RTP and operating moderator temperature, the excess positive reactivity is compensated by burnable absorbers (e.g., gadolinia), control rods, and whatever neutron

(continued)



BASES

BACKGROUND
(continued)

poisons (mainly xenon and samarium) are present in the fuel. The predicted core reactivity, as represented by control rod density, is calculated by a 3D core simulator code as a function of cycle exposure. This calculation is performed for projected operating states and conditions throughout the cycle. The core reactivity is determined from control rod densities for actual plant conditions and is then compared to the predicted value for the cycle exposure.

APPLICABLE
SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations (Ref. 2). In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod drop accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity:

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted rod density for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict rod density may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured rod density from the predicted rod density that develop during fuel depletion may be an indication that the assumptions of the DBA and transient analyses are no longer valid, or that an unexpected change in core conditions has occurred.

Reactivity anomalies satisfy Criterion 2 of the NRC Policy Statement (Ref. 3).

(continued)



BASES (continued)

LCO

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the "Nuclear Design Methodology" are larger than expected. A limit on the difference between the monitored and the predicted rod density corresponding to a reactivity difference of $\pm 1\% \Delta k/k$ has been established based on engineering judgment. A $> 1\%$ deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

APPLICABILITY

In MODE 1, most of the control rods are withdrawn and steady state operation is typically achieved. Under these conditions, the comparison between predicted and monitored core reactivity provides an effective measure of the reactivity anomaly. In MODE 2, control rods are typically being withdrawn during a startup. In MODES 3 and 4, all control rods are fully inserted and therefore the reactor is in the least reactive state, where monitoring core reactivity is not necessary. In MODE 5, fuel loading results in a continually changing core reactivity. SDM requirements (LCO 3.1.1) ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, shuffling). The SDM test, required by LCO 3.1.1, provides a direct comparison of the predicted and monitored core reactivity at cold conditions; therefore, the reactivity anomaly LCO is not applicable during these conditions.

ACTIONS

A.1

Should an anomaly develop between actual and expected critical rod configuration, the core reactivity difference must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This

(continued)



BASES

ACTIONS

A.1 (continued)

evaluation normally reviews the core conditions to determine their consistency with input to design calculations. Measured core and process parameters are also normally evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models may be reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ 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 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.2.1

Verifying the reactivity difference between the actual critical rod configuration and the expected configuration is within the limits of the LCO provides added assurance that plant operation is maintained within the assumptions of the DBA and transient analyses. The core monitoring software calculates the k-effective for the critical rod configuration and reactor conditions. A comparison of this calculated k-effective at the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. Control rod

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.2.1 (continued)

replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the monitored and predicted rod density can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at $\geq 75\%$ RTP have been obtained. The 1000 MWD/T Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
 2. FSAR, Chapter 14.6.
 3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Control Rod OPERABILITY

BASES

BACKGROUND

Control rods are components of the control rod drive (CRD) System, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure under conditions of normal operation, including abnormal operational transients, that specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements of GDC 26, GDC 27, GDC 28, and GDC 29 (Ref. 1).

The CRD System consists of 185 locking piston control rod drive mechanisms (CRDMs) and a hydraulic control unit for each drive mechanism. The locking piston type CRDM is a double acting hydraulic piston, which uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion.

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2, 3, and 4.

APPLICABLE
SAFETY ANALYSES

The analytical methods and assumptions used in the evaluations involving control rods are presented in References 2, 3, and 4. The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical and for limiting the potential effects of reactivity insertion events caused by malfunctions in the CRD System.

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The capability to insert the control rods provides assurance that the assumptions for scram reactivity in the DBA and transient analyses are not violated. Since the SDM ensures the reactor will be subcritical with the highest worth control rod withdrawn (assumed single failure), the additional failure of a second control rod to insert, if required, could invalidate the demonstrated SDM and potentially limit the ability of the CRD System to hold the reactor subcritical. If the control rod is stuck at an inserted position and becomes decoupled from the CRD, a control rod drop accident (CRDA) can possibly occur. Therefore, the requirement that all control rods be OPERABLE ensures the CRD System can perform its intended function.

The control rods also protect the fuel from damage which could result in release of radioactivity. The limits protected are the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs" and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), and the fuel damage limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events.

The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against fuel damage limits during a CRDA. The Bases for LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6 discuss in more detail how the SLs are protected by the CRD System.

Control rod OPERABILITY satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

LCO

The OPERABILITY of an individual control rod is based on a combination of factors, primarily, the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram accumulator status for a control rod only affects the scram insertion times; therefore, an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to

(continued)



BASES

LCO
(continued)

satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

APPLICABILITY

In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Control Rod OPERABILITY-Refueling."

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each control rod. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable control rod. Complying with the Required Actions may allow for continued operation, and subsequent inoperable control rods are governed by subsequent Condition entry and application of associated Required Actions.

A.1, A.2, A.3, and A.4

A control rod is considered stuck if it will not insert by either CRD drive water or scram pressure. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Actions are modified by a Note, which allows the rod worth minimizer (RWM) to be bypassed if required to allow continued operation. LCO 3.3.2.1, "Control Rod Block Instrumentation," provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, the local scram reactivity rate assumptions may not be met if the stuck control rod separation criteria are not met. Therefore, a verification that the separation criteria are met must be performed immediately. The separation criteria are not met if a) the stuck control rod occupies a location adjacent to

(continued)



BASES

ACTIONS

A.1, A.2, A.3, and A.4 (continued)

two "slow" control rods, b) the stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods adjacent to one another. The description of "slow" control rod is provided in LCO 3.1.4, "Control Rod Scram Times." In addition, the associated control rod drive must be disarmed in 2 hours.

Hydraulically disarming does not normally include isolation of the cooling water. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable time to perform the Required Action in an orderly manner. The control rod must be isolated from both scram and normal insert and withdraw pressure. Isolating the control rod from scram prevents damage to the CRDM.

Monitoring of the insertion capability of each withdrawn control rod must also be performed within 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM. SR 3.1.3.2 and SR 3.1.3.3 perform periodic tests of the control rod insertion capability of withdrawn control rods. Testing each withdrawn control rod ensures that a generic problem does not exist. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." The Required Action A.3 Completion Time only begins upon discovery that THERMAL POWER is greater than the actual LPSP of the RWM since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The allowed Completion Time of 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the LPSP of the RWM provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests.

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to

(continued)



BASES

ACTIONS

A.1, A.2, A.3 and A.4 (continued)

preserve the single failure criterion, an additional control rod would have to be assumed to fail to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

The allowed Completion Time of 72 hours to verify SDM is adequate, considering that with a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram.

B.1 and B.2

With two or more withdrawn control rods stuck, the stuck control rods must be isolated from scram pressure within 2 hours and the plant brought to MODE 3 within 12 hours. The control rod must be isolated from both scram and normal insert and withdraw pressure. Isolating the control rod from scram prevents damage to the CRDM. The allowed Completion Time is acceptable, considering the low probability of a CRDA occurring during this interval. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the control rods are fully inserted

(continued)



BASES

ACTIONS

C.1 and C.2 (continued)

within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed (electrically or hydraulically) to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves while maintaining cooling water to the CRD. The control rods can be electrically disarmed by disconnecting power from all four directional control valve solenoids. Required Action C.1 is modified by a Note, which allows the RWM to be bypassed if required to allow insertion of the inoperable control rods and continued operation. LCO 3.3.2.1 provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis.

The allowed Completion Times are reasonable, considering the small number of allowed inoperable control rods, and provide time to insert and disarm the control rods in an orderly manner and without challenging plant systems.

D.1 and D.2

Out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. At $\leq 10\%$ RTP, the generic banked position withdrawal sequence (BPWS) analysis (Ref. 5) requires inoperable control rods not in compliance with BPWS to be separated by at least two OPERABLE control rods in all directions, including the diagonal. Therefore, if two or more inoperable control rods are not in compliance with BPWS and not separated by at least two OPERABLE control rods, action must be taken to restore compliance with BPWS or restore the control rods to OPERABLE status. Condition D is modified by a Note indicating that the Condition is not applicable when $> 10\%$ RTP, since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.6. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring.

(continued)



BASES

ACTIONS
(continued)

E.1

If any Required Action and associated Completion Time of Condition A, C, or D are not met, or there are nine or more inoperable control rods, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. This ensures all insertable control rods are inserted and places the reactor in a condition that does not require the active function (i.e., scram) of the control rods. Below 10% RTP, the generic banked position withdrawal sequence (BPWS) analysis (Ref. 5) allows a maximum of eight bypassed control rods. The number of control rods permitted to be inoperable when operating above 10% RTP (e.g., no CRDA considerations) could be more than the value specified, but the occurrence of a large number of inoperable control rods could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.3.1

The position of each control rod must be determined to ensure adequate information on control rod position is available to the operator for determining control rod OPERABILITY and controlling rod patterns. Control rod position may be determined by the use of OPERABLE position indicators, by moving control rods to a position with an OPERABLE indicator, or by the use of other appropriate methods. The 24 hour Frequency of this SR is based on operating experience related to expected changes in control rod position and the availability of control rod position indications in the control room.

SR 3.1.3.2 and SR 3.1.3.3

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.3.2 and SR 3.1.3.3 (continued)

position. This ensures the control rod is not stuck and is free to insert on a scram signal. These Surveillances are not required when THERMAL POWER is less than or equal to the actual LPSP of the RWM, since the notch insertions may not be compatible with the requirements of banked position withdrawal sequence (BPWS) (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The 7 day Frequency of SR 3.1.3.2 is based on operating experience related to the changes in CRD performance and the ease of performing notch testing for fully withdrawn control rods. Partially withdrawn control rods are tested at a 31 day Frequency, based on the potential power reduction required to allow the control rod movement and considering the large testing sample of SR 3.1.3.2. Furthermore, the 31 day Frequency takes into account operating experience related to changes in CRD performance. At any time, if a control rod is immovable, a determination of that control rod's trippability must be made and appropriate action taken.

SR 3.1.3.4

Verifying that the scram time for each control rod to notch position 06 is ≤ 7 seconds provides reasonable assurance that the control rod will insert when required during a DBA or transient, thereby completing its shutdown function. This SR is performed in conjunction with the control rod scram time testing of SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," and the functional testing of SDV vent and drain valves in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlap this Surveillance to provide complete testing of the assumed safety function. The associated Frequencies are acceptable, considering the more frequent testing performed to demonstrate other aspects of control rod OPERABILITY and operating experience, which shows scram times do not significantly change over an operating cycle.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.1.3.5

Coupling verification is performed to ensure the control rod is connected to the CRDM and will perform its intended function when necessary. The Surveillance requires verifying a control rod does not go to the withdrawn overtravel position. The overtravel position feature provides a positive check on the coupling integrity since only an uncoupled CRD can reach the overtravel position. The verification is required to be performed any time a control rod is withdrawn to the "full out" position (notch position 48) or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This includes control rods inserted one notch and then returned to the "full out" position during the performance of SR 3.1.3.2. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved and operating experience related to uncoupling events.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 27, GDC 28, and GDC 29.
 2. FSAR, Section 3.4.6.
 3. FSAR, Section 14.5.
 4. FSAR, Section 14.6.
 5. NEDO-21231, "Banked Position Withdrawal Sequence," Section 7.2, January 1977.
 6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Scram Times

BASES

BACKGROUND

The scram function of the Control Rod Drive (CRD) System controls reactivity changes during abnormal operational transients to ensure that specified acceptable fuel design limits are not exceeded (Ref. 1). The control rods are scrammed by positive means using hydraulic pressure exerted on the CRD piston.

When a scram signal is initiated, control air is vented from the scram valves, allowing them to open by spring action. Opening the exhaust valve reduces the pressure above the main drive piston to atmospheric pressure, and opening the inlet valve applies the accumulator or reactor pressure to the bottom of the piston. Since the notches in the index tube are tapered on the lower edge, the collet fingers are forced open by cam action, allowing the index tube to move upward without restriction because of the high differential pressure across the piston. As the drive moves upward and the accumulator pressure reduces below the reactor pressure, a ball check valve opens, letting the reactor pressure complete the scram action. If the reactor pressure is low, such as during startup, the accumulator will fully insert the control rod in the required time without assistance from reactor pressure.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 2, 3, and 4. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. The resulting negative scram reactivity forms the basis for the determination of plant thermal limits (e.g., the MCPR). Other distributions of scram times (e.g., several control rods scrambling slower than the average time with several control rods scrambling faster than the average time) can also provide sufficient scram reactivity. Surveillance of each individual control rod's scram time ensures the scram reactivity assumed in the DBA and transient analyses can be met.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded. Above 800 psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL, during the analyzed limiting power transient. Below 800 psig, the scram function is assumed to perform during the control rod drop accident (Ref. 5) and, therefore, also provides protection against violating fuel damage limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis, the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits.

Control rod scram times satisfy Criterion 3 of the NRC Policy Statement (Ref. 7).

LCO

The scram times specified in Table 3.1.4-1 (in the accompanying LCO) are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met (Ref. 6).

To account for single failures and "slow" scrambling control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin that allows up to approximately 7% of the control rods (e.g., $185 \times 7\% \approx 13$) to have scram times exceeding the specified limits (i.e., "slow" control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens

(continued)



BASES

LCO
(continued)

("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished through measurement of the "dropout" times. To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed "slow" control rods may occupy adjacent locations.

Table 3.1.4-1 is modified by two Notes, which state that control rods with scram times not within the limits of the table are considered "slow" and that control rods with scram times > 7 seconds are considered inoperable as required by SR 3.1.3.4.

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed (LCO 3.1.3). Slow scrambling control rods can be conservatively declared inoperable and not accounted for as "slow" control rods.

APPLICABILITY

In MODES 1 and 2, a scram is assumed to function during transients and accidents analyzed for these plant conditions. These events are assumed to occur during startup and power operation; therefore, the scram function of the control rods is required during these MODES. In MODES 3 and 4, the control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram capability during these conditions. Scram requirements in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."

ACTIONS

A.1

When the requirements of this LCO are not met, the rate of negative reactivity insertion during a scram may not be within the assumptions of the safety analysis. Therefore, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

The four SRs of this LCO are modified by a Note stating that during a single control rod scram time surveillance, the CRD pumps shall be isolated from the associated scram accumulator. With the CRD pump isolated, (i.e., charging valve closed) the influence of the CRD pump head does not affect the single control rod scram times. During a full core scram, the CRD pump head would be seen by all control rods and would have a negligible effect on the scram insertion times.

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on an assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 800 psig demonstrates acceptable scram times for the transients analyzed in References 3 and 4.

Maximum scram insertion times occur at a reactor steam dome pressure of approximately 800 psig because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure ≥ 800 psig ensures that the measured scram times will be within the specified limits at higher pressures. To ensure that scram time testing is performed within a reasonable time following fuel movement within the reactor pressure vessel after a shutdown ≥ 120 days or longer, control rods are required to be tested before exceeding 40% RTP following the shutdown. The SR is modified by a Note stating that in the event fuel movement is limited to selected core cells, only those CRDs associated with the core cells affected by the fuel movements are required to be scram time tested. However, if the reactor remains shutdown ≥ 120 days, all control rods are required to be scram time tested. This Frequency is acceptable considering the additional surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected by work on control rods or the CRD System.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.4.2

Additional testing of a sample of control rods is required to verify the continued performance of the scram function during the cycle. A representative sample contains at least 10% of the control rods. This sample remains representative if no more than 20% of the control rods in the sample tested are determined to be "slow." With more than 20% of the sample declared to be "slow" per the criteria in Table 3.1.4-1, additional control rods are tested until this 20% criterion (i.e., 20% of the entire sample) is satisfied, or until the total number of "slow" control rods (throughout the core from all Surveillances) exceeds the LCO limit. For planned testing, the control rods selected for the sample should be different for each test. Data from inadvertent scrams should be used whenever possible to avoid unnecessary testing at power, even if the control rods with data may have been previously tested in a sample. The 120 day Frequency is based on operating experience that has shown control rod scram times do not significantly change over an operating cycle. This Frequency is also reasonable based on the additional Surveillances done on the CRDs at more frequent intervals in accordance with LCO 3.1.3 and LCO 3.1.5, "Control Rod Scram Accumulators."

SR 3.1.4.3

When work that could affect the scram insertion time is performed on a control rod or the CRD System, testing must be done to demonstrate that each affected control rod retains adequate scram performance over the range of applicable reactor pressures from zero to the maximum permissible pressure. The scram testing must be performed once before declaring the control rod OPERABLE. The required scram testing must demonstrate that for the affected control rod the scram valves open and the scram discharge path is open. This test can be performed with the control rod inserted and the accumulator drained and isolated to minimize potential damage to the drive. The test is adequate based on a high probability of meeting the scram time testing acceptance criteria at reactor pressures ≥ 800 psig. Limits for ≥ 800 psig are found in Table 3.1.4-1.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.4.3 (continued)

Specific examples of work that could affect the scram times are (but are not limited to) the following: removal of any CRD for maintenance or modification; replacement of a control rod; and maintenance or modification of a scram solenoid pilot valve, scram valve, accumulator, isolation valve or check valve in the piping required for scram.

The Frequency of once prior to declaring the affected control rod OPERABLE is acceptable because of the capability to test the control rod over a range of operating conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

SR 3.1.4.4

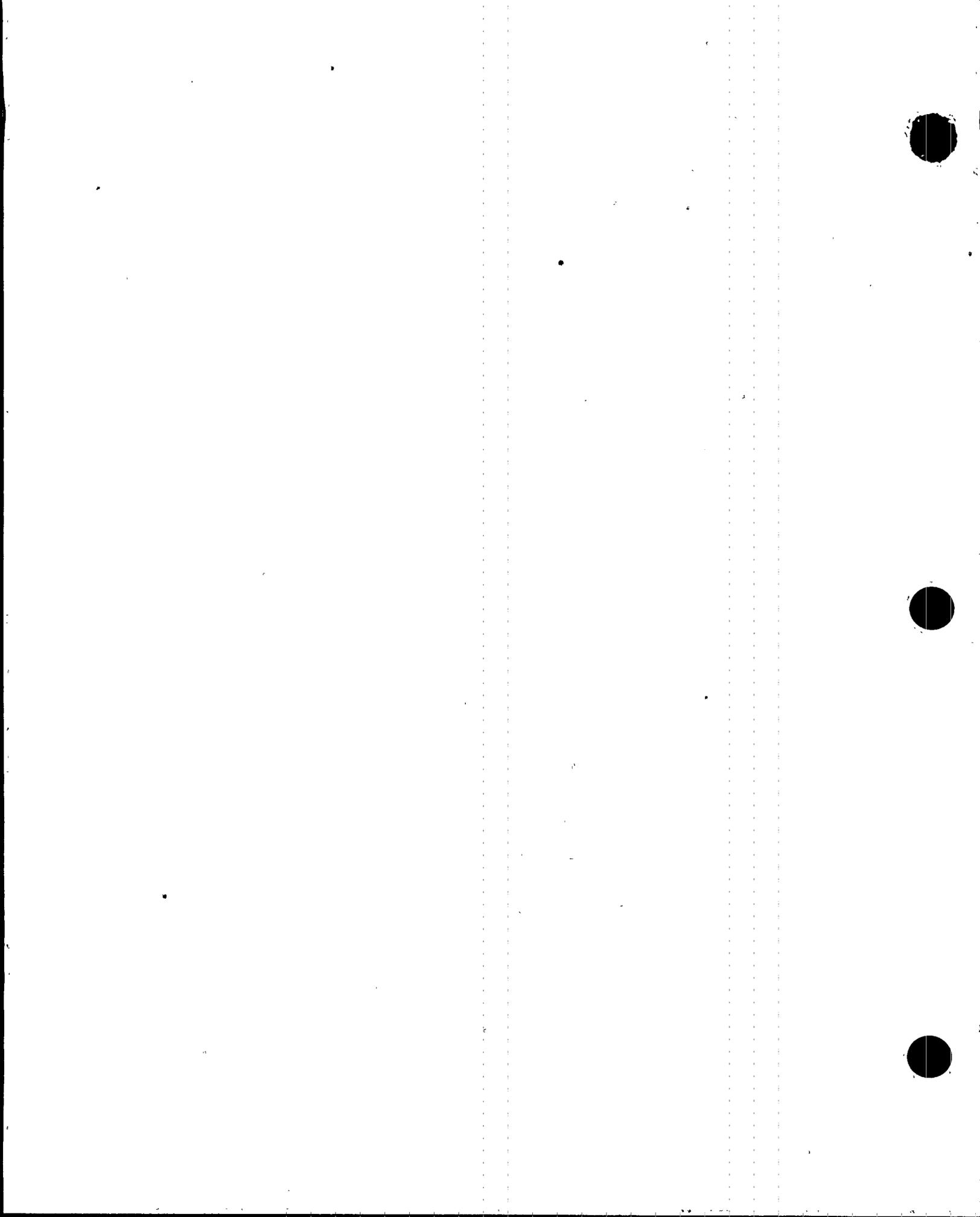
When work that could affect the scram insertion time is performed on a control rod or CRD System, testing must be done to demonstrate each affected control rod is still within the limits of Table 3.1.4-1 with the reactor steam dome pressure ≥ 800 psig. Where work has been performed at high reactor pressure, the requirements of SR 3.1.4.3 and SR 3.1.4.4 can be satisfied with one test. For a control rod affected by work performed while shut down, however, a zero pressure and high pressure test may be required. This testing ensures that, prior to withdrawing the control rod for continued operation, the control rod scram performance is acceptable for operating reactor pressure conditions. Alternatively, a control rod scram test during hydrostatic pressure testing could also satisfy both criteria.

The Frequency of once prior to exceeding 40% RTP is acceptable because of the capability to test the control rod over a range of operating conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section 3.4.6.
3. FSAR, Section 14.5.

(continued)



BASES

REFERENCES
(continued)

4. FSAR, Section 14.6.
5. NEDE-24011-P-A-11, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, November 1995.
6. Letter from R. F. Janecek (BWROG) to R. W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, September 17, 1987.
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.



B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Rod Scram Accumulators

BASES

BACKGROUND

The control rod scram accumulators are part of the Control Rod Drive (CRD) System and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen, which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.1.4, "Control Rod Scram Times."

APPLICABLE
SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 1, 2, and 3. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. OPERABILITY of each individual control rod scram accumulator, along with LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.4, ensures that the scram reactivity assumed in the DBA and transient analyses can be met. The existence of an inoperable accumulator may invalidate prior scram time measurements for the associated control rod.

The scram function of the CRD System, and therefore the OPERABILITY of the accumulators, protects the MCPR Safety Limit (see Bases for SL 2.1.1, "Reactor Core SLs" and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded (see Bases for LCO 3.1.4). In addition, the scram function at low reactor vessel pressure (i.e., startup conditions) provides protection against violating fuel design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control").

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Control rod scram accumulators satisfy Criterion 3 of the NRC Policy Statement (Ref. 4).

LCO

The OPERABILITY of the control rod scram accumulators is required to ensure that adequate scram insertion capability exists when needed over the entire range of reactor pressures. The OPERABILITY of the scram accumulators is based on maintaining adequate accumulator pressure.

APPLICABILITY

In MODES 1 and 2, the scram function is required for mitigation of DBAs and transients, and therefore the scram accumulators must be OPERABLE to support the scram function. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. Requirements for scram accumulators in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each control rod scram accumulator. This is acceptable since the Required Actions for each Condition provide appropriate compensatory actions for each affected accumulator. Complying with the Required Actions may allow for continued operation and subsequent affected accumulators governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

With one control rod scram accumulator inoperable and the reactor steam dome pressure ≥ 900 psig, the control rod may be declared "slow," since the control rod will still scram at the reactor operating pressure but may not satisfy the required scram times in Table 3.1.4-1.

(continued)



BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time test. Otherwise, the control rod would already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action A.2) and LCO 3.1.3 is entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function, in accordance with ACTIONS of LCO 3.1.3.

The allowed Completion Time of 8 hours is reasonable, based on the large number of control rods available to provide the scram function and the ability of the affected control rod to scram only with reactor pressure at high reactor pressures.

B.1, B.2.1, and B.2.2

With two or more control rod scram accumulators inoperable and reactor steam dome pressure ≥ 900 psig, adequate pressure must be supplied to the charging water header. With inadequate charging water pressure, all of the accumulators could become inoperable, resulting in a potentially severe degradation of the scram performance. Therefore, within 20 minutes from discovery of charging water header pressure < 940 psig concurrent with Condition B, adequate charging water header pressure must be restored. The allowed Completion Time of 20 minutes is reasonable, to place a CRD pump into service to restore the charging water header pressure, if required. This Completion Time is based on the ability of the reactor pressure alone to fully insert all control rods.

The control rod may be declared "slow," since the control rod will still scram using only reactor pressure, but may not satisfy the times in Table 3.1.4-1. Required Action B.2.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control scram time is within the limits of Table 3.1.4-1 during the last scram time test. Otherwise, the control rod

(continued)



BASES

ACTIONS

B.1, B.2.1, and B.2.2 (continued)

- would already be considered "slow" and the further degradation of scram performance with an inoperable accumulator could result in excessive scram times. In this event, the associated control rod is declared inoperable (Required Action B.2.2) and LCO 3.1.3 entered. This would result in requiring the affected control rod to be fully inserted and disarmed, thereby satisfying its intended function in accordance with ACTIONS of LCO 3.1.3.

The allowed Completion Time of 1 hour is reasonable, based on the ability of only the reactor pressure to scram the control rods and the low probability of a DBA or transient occurring while the affected accumulators are inoperable.

C.1 and C.2

With one or more control rod scram accumulators inoperable and the reactor steam dome pressure < 900 psig, the pressure supplied to the charging water header must be adequate to ensure that accumulators remain charged. With the reactor steam dome pressure < 900 psig, the function of the accumulators in providing the scram force becomes much more important since the scram function could become severely degraded during a depressurization event or at low reactor pressures. Therefore, immediately upon discovery of charging water header pressure < 940 psig, concurrent with Condition C, all control rods associated with inoperable accumulators must be verified to be fully inserted. Withdrawn control rods with inoperable accumulators may fail to scram under these low pressure conditions. The associated control rods must also be declared inoperable within 1 hour. The allowed Completion Time of 1 hour is reasonable for Required Action C.2, considering the low probability of a DBA or transient occurring during the time that the accumulator is inoperable.

D.1

The reactor mode switch must be immediately placed in the shutdown position if either Required Action and associated Completion Time associated with the loss of the CRD charging pump (Required Actions B.1 and C.1) cannot be met. This

(continued)



BASES

ACTIONS.

D.1 (continued)

ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. This Required Action is modified by a Note stating that the action is not applicable if all control rods associated with the inoperable scram accumulators are fully inserted, since the function of the control rods has been performed.

SURVEILLANCE
REQUIREMENTS

SR 3.1.5.1

SR 3.1.5.1 requires that the accumulator pressure be checked every 7 days to ensure adequate accumulator pressure exists to provide sufficient scram force. An automatic accumulator monitor may be used to continuously satisfy this requirement. The primary indicator of accumulator OPERABILITY is the accumulator pressure. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1100 psig (Ref. 1). Declaring the accumulator inoperable when the minimum pressure is not maintained ensures that significant degradation in scram times does not occur. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

REFERENCES

1. FSAR, Section 3.4.6.
 2. FSAR, Section 14.5.
 3. FSAR, Section 14.6.
 4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1 and 2. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 3), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Refs. 4 and 5). Generic evaluations (Refs. 1 and 6) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 7) and the calculated offsite doses will be well within the required limits (Ref. 5).

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. Generic analysis of the BPWS (Ref. 8) has demonstrated that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation. The evaluation provided by the generic BPWS analysis (Ref. 8) allows a limited number (i.e., eight) and corresponding distribution of fully inserted, inoperable control rods, that are not in compliance with the sequence.

Rod pattern control satisfies Criterion 3 of the NRC Policy Statement (Ref. 9).

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.

APPLICABILITY

In MODES 1 and 2, when THERMAL POWER is \leq 10% RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is $>$ 10% RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 2). In MODES 3, 4, and 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

(continued)



BASES (continued)

ACTIONS

A.1 and A.2

With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence (Ref. 8), actions may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to $\leq 10\%$ RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight, to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence. When the control rod pattern is not in compliance with the prescribed sequence, all control rod movement must be stopped except for moves needed to correct the rod pattern, or scram if warranted.

Required Action A.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator or a qualified member of the technical staff. This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence (Ref. 8). Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of

(continued)



BASES

ACTIONS

B.1 and B.2 (continued)

control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position.

LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator or a qualified member of the technical staff.

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE
REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at $\leq 10\%$ RTP.

REFERENCES

1. NEDE-24011-P-A-11-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States," Section 2.2.3.1, November 1995.
2. Letter from T. Pickens (BWROG) to G. C. Lainas (NRC), Amendment 17 to General Electric Licensing Topical Report, NEDE-24011-P-A, August 15, 1986.
3. NUREG-0979, Section 4.2.1.3.2, April 1983.
4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

(continued)



BASES

REFERENCES
(continued)

5. 10 CFR 100.11.
 6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.
 7. ASME, Boiler and Pressure Vessel Code.
 8. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.
 9. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram.

The SLC System consists of a boron solution storage tank, two positive displacement pumps in parallel and two explosive valves in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

The worst case sodium pentaborate solution concentration required to shutdown the reactor with sufficient margin to account for $0.05 \Delta k/k$ and Xenon poisoning effects is 9.2 weight percent. This corresponds to a 40°F saturation temperature. The worst case SLCS equipment area temperature is not predicted to fall below 50°F. This provides a 10°F thermal margin to unwanted precipitation of the sodium pentaborate. Tank heating components provide backup assurance that the sodium pentaborate solution temperature will never fall below 50°F but are not required for TS operability considerations.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating instructions, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 660 ppm of natural boron, in the reactor coolant at 70°F. To allow for imperfect mixing, leakage and the volume in other piping connected to the reactor system, an amount of boron equal to 25% of the amount cited above is added (Ref. 2). This volume versus concentration limit and the temperature versus concentration limits in Figure 3.1.7-1 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in the entire residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

The SLC System satisfies Criterion 4 of the NRC Policy Statement (Ref. 3).

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5,

(continued)



BASES

APPLICABILITY
(continued)

only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

ACTIONS

A.1

If one SLC subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the shutdown function. However, the overall reliability is reduced because a single failure in the remaining OPERABLE subsystem could result in reduced SLC System shutdown capability. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the intended SLC System function and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive (CRD) System to shut down the plant.

B.1

If both SLC subsystems are inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor.

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.7.1

SR 3.1.7.1 is a 24 hour Surveillance verifying the volume of the borated solution in the storage tank, thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. This Surveillance ensures that the proper borated solution volume is maintained. The sodium pentaborate solution concentration requirements ($\leq 9.2\%$ by weight) and the required quantity of Boron-10 (≥ 186 lbs) establish the tank volume requirement. The 24 hour Frequency is based on operating experience that has shown there are relatively slow variations in the solution volume.

SR 3.1.7.2

SR 3.1.7.2 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. An automatic continuity monitor may be used to continuously satisfy this requirement. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.3 and SR 3.1.7.5

SR 3.1.7.3 requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank. The concentration is dependent upon the volume of water and quantity of boron in the storage tank. SR 3.1.7.5 requires verification that the SLC system conditions satisfy the following equation:

$$\frac{(C)(Q)(E)}{(13 \text{ WT } \%)(86 \text{ GPM})(19.8 \text{ ATOM } \%)} > = 1.0$$

C = sodium pentaborate solution weight percent concentration

Q = SLC system pump flow rate in gpm

E = Boron-10 atom percent enrichment in the sodium pentaborate solution

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.7.3 and SR 3.1.7.5 (continued)

To meet 10 CFR 50.62, the SLC System must have a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent natural sodium pentaborate solution. The atom percentage of natural B-10 is 19.8%. This equivalency requirement is met when the equation given above is satisfied. The equation can be satisfied by adjusting the solution concentration, pump flow rate or Boron-10 enrichment. If the results of the equation are < 1 , the SLC System is no longer capable of shutting down the reactor with the margin described in Reference 2. However, the quantity of stored boron includes an additional margin (25%) beyond the amount needed to shut down the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water, leakage, and the volume in other piping connected to the reactor system.

The sodium pentaborate solution (SPB) concentration is allowed to be > 9.2 weight percent provided the concentration and temperature of the sodium pentaborate solution are verified to be within the limits of Figure 3.1.7-1. This ensures that unwanted precipitation of the sodium pentaborate does not occur.

SR 3.1.7.3 and SR 3.1.7.5 must be performed every 31 days or within 24 hours of when boron or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. The 31 day Frequency of these Surveillances is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.3 must be performed within 8 hours of discovery that the concentration is > 9.2 weight percent and every 12 hours thereafter until the concentration is verified to be ≤ 9.2 weight percent. This Frequency is appropriate under these conditions taking into consideration the SLC System design capability still exists for vessel injection under these conditions and the low probability of the temperature and concentration limits of Figure 3.1.7-1 not being met.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.1.7.4

This Surveillance requires the amount of Boron-10 in the SLC solution tank to be determined every 31 days. The enriched sodium pentaborate solution is made by combining stoichiometric quantities of borax and boric acid in demineralized water. Since the chemicals used have known Boron-10 quantities, the Boron-10 quantity in the sodium pentaborate solution formed can be calculated. This parameter is used as input to determine the volume requirements for SR 3.1.7.1. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.6

Demonstrating that each SLC System pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ 1275 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration and enrichment requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. The 18 month Frequency is acceptable since inservice testing of the pumps, performed every 92 days, will detect any adverse trends in pump performance.

SR 3.1.7.7 and SR 3.1.7.8

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 36 months at alternating 18 month intervals. The Surveillance may be performed in separate steps to

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.7.7 and SR 3.1.7.8 (continued)

prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all piping between the boron solution storage tank and the suction inlet to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping is unblocked is to pump from the storage tank to the storage tank. The 18 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the piping or by other means.

SR 3.1.7.9

The enriched sodium pentaborate solution is made by combining stoichiometric quantities of borax and boric acid in demineralized water. Isotopic tests on these chemicals to verify the actual B-10 enrichment must be performed at least every 18 months and after addition of boron to the SLC tank in order to ensure that the proper B-10 atom percentage is being used and SR 3.1.7.5 will be met. The sodium pentaborate enrichment must be calculated within 24 hours and verified by analysis within 30 days.

REFERENCES

1. 10 CFR 50.62.
 2. FSAR, Section 3.8.4.
 3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. Each instrument volume is connected to the radwaste system by a drain line containing two valves in series. Each header is connected to a common vent line with two valves in series for a total of four vent valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE
SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scrambling. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

- a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 3); and
- b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. The offsite doses resulting from reactor coolant discharge from the SDV are significantly lower than the bounding doses resulting from a main steam line break outside the secondary containment (Ref. 2) and are well within the limits of

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

10 CFR 100 (Ref. 3). Adequate core cooling is by the integrated operation of the Emergency Core Cooling Systems (Ref. 4). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in the instrumentation volume exceeds a specified setpoint. The setpoint is chosen so that all control rods are inserted before the SDV has insufficient volume to accept a full scram.

SDV vent and drain valves satisfy Criterion 3 of the NRC Policy Statement (Ref. 5).

LCO

The OPERABILITY of all SDV vent and drain valves ensures that the SDV vent and drain valves will close during a scram to contain reactor water discharged to the SDV piping. Since each vent and drain line is provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to open on scram reset to ensure that a path is available for the SDV piping to drain freely at other times.

APPLICABILITY

In MODES 1 and 2, scram may be required; therefore, the SDV vent and drain valves must be OPERABLE. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that only a single control rod can be withdrawn. Also, during MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Therefore, the SDV vent and drain valves are not required to be OPERABLE in these MODES since the reactor is subcritical and only one rod may be withdrawn and subject to scram.

(continued)



BASES

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each SDV vent and drain line. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SDV line. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SDV lines are governed by subsequent Condition entry and application of associated Required Actions.

A.1

When one SDV vent or drain valve is inoperable in one or more lines, the valve must be restored to OPERABLE status within 7 days. The Completion Time is reasonable, given the level of redundancy in the lines and the low probability of a scram occurring during the time the valve(s) are inoperable. The SDV is still isolable since the redundant valve in the affected line is OPERABLE. During these periods, the single failure criterion may not be preserved, and a higher risk exists to allow reactor water out of the primary system during a scram.

B.1

If both valves in a line are inoperable, the line must be isolated to contain the reactor coolant during a scram. When a line is isolated, the potential for an inadvertent scram due to high SDV level is increased. Required Action B.1 is modified by a Note that allows periodic draining and venting of the SDV when a line is isolated. During these periods, the line may be unisolated under administrative control. This allows any accumulated water in the line to be drained, to preclude a reactor scram on SDV high level. This is acceptable since the administrative controls ensure the valve can be closed quickly, by a dedicated operator, if a scram occurs with the valve open.

The 8 hour Completion Time to isolate the line is based on the low probability of a scram occurring while the line is not isolated and unlikelihood of significant CRD seal leakage.

(continued)



BASES

ACTIONS
(continued)

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.8.1

During normal operation, the SDV vent and drain valves should be in the open position (except when performing SR 3.1.8.2) to allow for drainage of the SDV piping. Verifying that each valve is in the open position ensures that the SDV vent and drain valves will perform their intended functions during normal operation. This SR does not require any testing or valve manipulation; rather, it involves verification that the valves are in the correct position.

The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation, which ensure correct valve positions.

SR 3.1.8.2

During a scram, the SDV vent and drain valves should close to contain the reactor water discharged to the SDV piping. Cycling each valve through its complete range of motion (closed and open) ensures that the valve will function properly during a scram. The 92 day Frequency is based on operating experience and takes into account the level of redundancy in the system design.

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.1.8.3

SR 3.1.8.3 is an integrated test of the SDV vent and drain valves to verify total system performance. After receipt of a simulated or actual scram signal, the closure of the SDV vent and drain valves is verified. The closure time of 60 seconds after receipt of a scram signal is acceptable based on the bounding analysis for release of reactor coolant outside containment (Ref. 2). Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3 overlap this Surveillance to provide complete testing of the assumed safety function. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 3.4.5.3.1.
 2. FSAR, Section 14.6.5.
 3. 10 CFR 100.
 4. FSAR, Section 6.5.
 5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, and 4.

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during abnormal operational transients for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure and fuel bundle type.

LOCA analyses are then performed to ensure that the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference 5. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).

LCO

The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses.

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 4) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels \leq 25% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

(continued)



BASES (continued)

ACTIONS
(continued)

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to $< 25\%$ RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to $< 25\%$ RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is $\geq 25\%$ RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER $\geq 25\%$ RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A-11 "General Electric Standard Application for Reactor Fuel," November 1995.
 2. FSAR, Chapter 3.
 3. FSAR, Chapter 14.
 4. FSAR, Appendix N.
 5. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 1, February 1996.
 6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during abnormal operational transients. Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 2, 3, 4, and 5. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (Δ CPR). When the largest Δ CPR is added to the MCPR SL, the required operating limit MCPR is obtained.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Flow dependent correction factor for MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 6) to analyze slow flow runout transients. The flow dependent correction factor is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 7).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the

(continued)



BASES

APPLICABILITY
(continued)

nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

B.1

If the MCPR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

(continued)



BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is \geq 25% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER \geq 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of τ , which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4, "Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter τ must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1 and SR 3.1.4.2 because the effective scram speed distribution may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle.

REFERENCES

1. NUREG-0562, "Fuel Rod Failure As a Consequence of Departure from Nucleate Boiling or Dryout," June 1979.
2. NEDE-24011-P-A-11, "General Electric Standard Application for Reactor Fuel," November 1995.
3. FSAR, Chapter 3.
4. FSAR, Chapter 14.

(continued)



BASES

REFERENCES
(continued)

5. FSAR, Appendix N.
 6. NEDO-30130-A, "Steady State Nuclear Methods,"
May 1985.
 7. NRC No. 93-102, "Final Policy Statement on Technical
Specification Improvements," July 23, 1993.
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including abnormal operational transients. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the anticipated operating conditions identified in Reference 1.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1 and 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

- a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO₂ pellet; and
- b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for abnormal operational transients, plus an allowance for densification power spiking.

The LHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 4).

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.

APPLICABILITY

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 25% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at $\geq 25\%$ RTP.

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limits such that the plant is operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LHGR out of specification.

B.1

If the LHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the

(continued)



BASES

ACTIONS

B.1 (continued)

LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER TO < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.2.3.1

The LHGR is required to be initially calculated within 12 hours after THERMAL POWER is $\geq 25\%$ RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slow changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER $\geq 25\%$ RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels.

REFERENCES

1. FSAR, Chapter 14.
 2. FSAR, Chapter 3.
 3. NUREG-0800, Standard Review Plan 4.2, Section II.A.2(g), Revision 2, July 1981.
 4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design," GDC 13, "Instrumentation and Control," GDC 20, "Protection System Functions," and GDC 23, "Protection System Failure Modes" (Ref. 1). This LCO is provided to require the APRM gain or APRM flow biased scram setpoints to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (F RTP), where F RTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

$$\frac{\text{MFLPD}}{\text{F RTP}} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by a gain adjustment on the APRMs or adjustment of the APRM setpoints. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to F RTP and thus maintains RTP margins for APLHGR and MCPR.

The normally selected APRM setpoints position the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The setpoints are flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow,

(continued)



BASES

BACKGROUND
(continued)

the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM setpoints are supported by the analyses presented in References 1 and 2 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR and MCPR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the flow biased APRM scram setpoints may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

The APRM neutron flux signal is also adjusted to more closely follow the fuel cladding heat flux during power transients. The APRM neutron flux signal is a measure of the core thermal power during steady state operation. During power transients, the APRM signal leads the actual core thermal power response because of the fuel thermal time constant. Therefore, on power increase transients, the APRM signal provides a conservatively high measure of core thermal power. By passing the APRM signal through an electronic filter with a time constant less than, but approximately equal to, that of the fuel thermal time constant, an APRM transient response that more closely follows actual fuel cladding heat flux is obtained, while a conservative margin is maintained. The delayed response of the filtered APRM signal allows the flow biased APRM scram levels to be positioned closer to the upper bound of the normal power and flow range, without unnecessarily causing reactor scrams during short duration neutron flux spikes. These spikes can be caused by insignificant transients such as performance of main steam line valve surveillances or momentary flow increases of only several percent.

APPLICABLE
SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR and MCPR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

FSAR safety analyses (Refs. 2 and 3) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR and MCPR) occurs.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of either the APLHGR or the MCPR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the SLs could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the FRTP, or the flow biased APRM scram level is required to be reduced by the ratio of FRTP to the core limiting MFLPD. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the flow biased APRM scram setpoints, dependent on the increased peaking that may be encountered.

The APRM gain and setpoints satisfy Criteria 2 and 3 of the NRC Policy Statement (Ref. 4).

LCO

Meeting any one of the following conditions ensures acceptable operating margins for events described above:

- a. Limiting excess power peaking;
- b. Reducing the APRM flow biased neutron flux upscale scram setpoints by multiplying the APRM setpoints by the ratio of FRTP and the core limiting value of MFLPD; or

(continued)



BASES

LCO
(continued)

- c. Increasing APRM gains to cause the APRM to read ≥ 100 times MFLPD (in %). This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM flow biased scram setpoints are reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM flow biased scram setpoints. Adjusting APRM gain or setpoints is equivalent to MFLPD less than or equal to FRTP, as stated in the LCO.

For compliance with LCO Item b (APRM setpoint adjustment) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are required to be adjusted. In addition, each APRM may be allowed to have its gain or setpoints adjusted independently of other APRMs that are having their gain or setpoints adjusted.

APPLICABILITY

The MFLPD limit, APRM gain adjustment, and APRM flow biased scram and associated setpoints are provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the reactor is operating at $\geq 25\%$ RTP.

ACTIONS

A.1

If the APRM gain or setpoints are not within limits while the MFLPD has exceeded FRTP, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit

(continued)



BASES

ACTIONS

A.1 (continued)

may be reduced. Therefore, prompt action should be taken to restore the MFLPD to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or setpoints to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

B.1

If MFLPD cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

The MFLPD is required to be calculated and compared with F RTP, or APRM gains or setpoint, to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are only required to determine the MFLPD and, assuming MFLPD is greater than F RTP, the appropriate gain or setpoint, and are not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram circuitry. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER \geq 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

The 12 hour Frequency of SR 3.2.4.2 requires a more frequent verification than if MFLPD is less than or equal to FRP. When MFLPD is greater than FRP, more rapid changes in power distribution are typically expected.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 23.
 2. FSAR, Chapter 14.
 3. FSAR, Chapter 3.
 4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.3 INSTRUMENTATION

B 3.3.1.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor scram when one or more monitored parameters exceed their specified limits, to preserve the integrity of the fuel cladding and the Reactor Coolant System (RCS) and minimize the energy that must be absorbed following a loss of coolant accident (LOCA). This can be accomplished either automatically or manually.

The protection and monitoring functions of the RPS have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance. The LSSS are defined in this Specification as the Allowable Values, which, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits, including Safety Limits (SLs) during Design Basis Accidents (DBAs).

The RPS, as described in the FSAR, Section 7.2 (Ref. 1), includes sensors, relays, bypass circuits, and switches that are necessary to cause initiation of a reactor scram. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. The input parameters to the scram logic are from instrumentation that monitors reactor vessel water level, reactor vessel pressure, neutron flux, main steam line isolation valve position, turbine control valve (TCV) fast closure (indicated by TCV low hydraulic pressure) trip oil pressure, turbine stop valve (TSV) position, drywell pressure, scram pilot air header pressure, and scram discharge volume (SDV) water level, as well as reactor mode switch in shutdown position, manual, and RPS channel test switch scram signals. There are at least four redundant sensor input signals from each of these parameters (with the exception of the reactor mode switch in shutdown and manual scram signals). Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay deenergizes actuators, which then outputs an RPS trip signal to the trip logic.

(continued)



BASES

BACKGROUND
(continued)

The RPS is comprised of two independent trip systems (A and B) with two logic channels in each trip system (logic channels A1 and A2, B1 and B2) as shown in Reference 1. The outputs of the logic channels in a trip system are combined in a one-out-of-two logic so that either channel can trip the associated trip system. The tripping of both trip systems will produce a reactor scram. This logic arrangement is referred to as a one-out-of-two taken twice logic. Each trip system can be reset by use of a reset switch. If a full scram occurs (both trip systems trip), a relay prevents reset of the trip systems for 10 seconds after the full scram signal is received. This 10 second delay on reset ensures that the scram function will be completed.

Two scram pilot valves are located in the hydraulic control unit for each control rod drive (CRD). Each scram pilot valve is solenoid operated, with the solenoids normally energized. The scram pilot valves control the air supply to the scram inlet and outlet valves for the associated CRD. When either scram pilot valve solenoid is energized, air pressure holds the scram valves closed and, therefore, both scram pilot valve solenoids must be de-energized to cause a control rod to scram. The scram valves control the supply and discharge paths for the CRD water during a scram. One of the scram pilot valve solenoids for each CRD is controlled by trip system A, and the other solenoid is controlled by trip system B. Any trip of trip system A in conjunction with any trip in trip system B results in de-energizing both solenoids, air bleeding off, scram valves opening, and control rod scram.

The backup scram valves, which energize on a full scram signal to depressurize the scram air header, are also controlled by the RPS. Additionally, the RPS System controls the SDV vent and drain valves such that when both trip systems trip, the SDV vent and drain valves close to isolate the SDV.

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The actions of the RPS are assumed in the safety analyses of References 1, 2, and 3. The RPS initiates a reactor scram when monitored parameter values exceed the Allowable Values, specified by the setpoint methodology and listed in Table 3.3.1.1-1 to preserve the integrity of the fuel

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

cladding, the reactor coolant pressure boundary (RCPB), and the containment by minimizing the energy that must be absorbed following a LOCA.

RPS instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 10). Functions not specifically credited in the accident analysis are retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The OPERABILITY of the RPS is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.1.1-1. Each Function must have a required number of OPERABLE channels per RPS trip system, with their setpoints within the specified Allowable Value, where appropriate. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint).

Allowable Values are specified for each RPS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The OPERABILITY of scram pilot valves and associated solenoids, backup scram valves, and SDV valves, described in the Background section, are not addressed by this LCO.

The individual Functions are required to be OPERABLE in the MODES or other specified conditions in the Table, which may require an RPS trip to mitigate the consequences of a design basis accident or transient. To ensure a reliable scram function, a combination of Functions are required in each MODE to provide primary and diverse initiation signals.

The only MODES specified in Table 3.3.1.1-1 are MODES 1 (which encompasses $\geq 30\%$ RTP) and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. No RPS Function is required in MODES 3 and 4 since all control rods are fully inserted and the Reactor Mode Switch Shutdown Position control rod withdrawal block (LCO 3.3.2.1) does not allow any control rod to be withdrawn. In MODE 5, control rods withdrawn from a core cell containing no fuel assemblies do not affect the reactivity of the core and, therefore, are not required to have the capability to scram. Provided all other control rods remain inserted, no RPS function is required. In this condition, the required SDM (LCO 3.1.1) and refuel position one-rod-out interlock (LCO 3.9.2) ensure that no event requiring RPS will occur.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Intermediate Range Monitor (IRM)

1.a. Intermediate Range Monitor Neutron Flux—High

The IRMs monitor neutron flux levels from the upper range of the source range monitor (SRM) to the lower range of the average power range monitors (APRMs). The IRMs are capable of generating trip signals that can be used to prevent fuel damage resulting from abnormal operating transients in the intermediate power range. In this power range, the most

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1.a. Intermediate Range Monitor Neutron Flux—High
(continued)

significant source of reactivity change is due to control rod withdrawal. The IRM mitigates control rod withdrawal error events and is diverse from the rod worth minimizer (RWM), which monitors and controls the movement of control rods at low power. The RWM prevents the withdrawal of an out of sequence control rod during startup that could result in an unacceptable neutron flux excursion (Ref. 2). The IRM provides mitigation of the neutron flux excursion. To demonstrate the capability of the IRM System to mitigate control rod withdrawal events, generic analyses have been performed (Ref. 3) to evaluate the consequences of control rod withdrawal events during startup that are mitigated only by the IRM. This analysis, which assumes that one IRM channel in each trip system is bypassed, demonstrates that the IRMs provide protection against local control rod withdrawal errors and results in peak fuel energy depositions below the 170 cal/gm fuel failure threshold criterion.

The IRMs are also capable of limiting other reactivity excursions during startup, such as cold water injection events; although no credit is specifically assumed.

The IRM System is divided into two groups of IRM channels, with four IRM channels inputting to each trip system. The analysis of Reference 3 assumes that one channel in each trip system is bypassed. Therefore, six channels with three channels in each trip system are required for IRM OPERABILITY to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This trip is active in each of the 10 ranges of the IRM, which must be selected by the operator to maintain the neutron flux within the monitored level of an IRM range.

The analysis of Reference 3 has adequate conservatism to permit an IRM Allowable Value of 120 divisions of a 125 division scale.

The Intermediate Range Monitor Neutron Flux—High Function must be OPERABLE during MODE 2 when control rods may be withdrawn and the potential for criticality exists. In MODE 5, when a cell with fuel has its control rod withdrawn, the IRMs provide monitoring for and protection against

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1.a. Intermediate Range Monitor Neutron Flux—High
(continued)

unexpected reactivity excursions. In MODE 1, the APRM System and the RBM provide protection against control rod withdrawal error events and the IRMs are not required.

1.b. Intermediate Range Monitor—Inop

This trip signal provides assurance that a minimum number of IRMs are OPERABLE. Anytime an IRM mode switch is moved to any position other than "Operate," the detector voltage drops below a preset level, or when a module is not plugged in, an inoperative trip signal will be received by the RPS unless the IRM is bypassed. Since only one IRM in each trip system may be bypassed, only one IRM in each RPS trip system may be inoperable without resulting in an RPS trip signal.

This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

Six channels of Intermediate Range Monitor—Inop with three channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

Since this Function is not assumed in the safety analysis, there is no Allowable Value for this Function.

This Function is required to be OPERABLE when the Intermediate Range Monitor Neutron Flux—High Function is required.

Average Power Range Monitor

2.a. Average Power Range Monitor Neutron Flux—High, Setdown

The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP. For operation at

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux—High,
Setdown (continued)

low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux—High, Setdown Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux—High, Setdown Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux—High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux—High, Setdown Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux—High, Setdown Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Neutron Flux—High, Setdown with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux—High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY2.a. Average Power Range Monitor Neutron Flux—High,
Setdown (continued)

In MODE 1, the Average Power Range Monitor Neutron Flux—High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Flow Biased Simulated
Thermal Power—High

The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than or equal to the Average Power Range Monitor Fixed Neutron Flux—High Function Allowable Value. The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event) and protects the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. During these events, the THERMAL POWER increase does not significantly lag the neutron flux response and, because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Fixed Neutron Flux—High Function will provide a scram signal before the Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function setpoint is exceeded.

The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated
Thermal Power-High (continued)

Average Power Range Monitor Flow Biased Simulated Thermal Power-High with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located. Each APRM channel receives a total drive flow signal representative of total core flow. The total drive flow signals are generated by two flow units, one of which supplies signals to the trip system A APRMs, while the other one supplies signals to the trip system B APRMs. Each flow unit signal is provided by summing up the flow signals from the two recirculation loops. Each required Average Power Range Monitor Flow Biased Simulated Thermal Power-High channel requires an input from its associated OPERABLE flow unit.

The clamped Allowable Value is based on analyses that take credit for the Average Power Range Monitor Flow Biased Simulated Thermal Power-High Function for the mitigation of the loss of feedwater heating event. The THERMAL POWER time constant of < 7 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER. The term "W" in the equation for determining the Allowable Value is defined as total recirculation flow in percent of rated.

The Average Power Range Monitor Flow Biased Simulated Thermal Power-High Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

2.c. Average Power Range Monitor Fixed Neutron Flux-High

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The Average Power Range Monitor Fixed Neutron Flux-High Function is capable of generating a

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

2.c. Average Power Range Monitor Fixed Neutron Flux—High
(continued)

trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux—High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 5) takes credit for the Average Power Range Monitor Fixed Neutron Flux—High Function to terminate the CRDA.

The APRM System is divided into two groups of channels with three APRM channels inputting to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Fixed Neutron Flux—High with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Fixed Neutron Flux—High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Fixed Neutron Flux—High Function is assumed in the CRDA analysis, which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux—High, Setdown Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor Fixed Neutron Flux—High Function is not required in MODE 2.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

2.d. Average Power Range Monitor—Downscale

This signal ensures that there is adequate Neutron Monitoring System protection if the reactor mode switch is placed in the run position prior to the APRMs coming on scale. With the reactor mode switch in run, an APRM downscale signal coincident with an associated Intermediate Range Monitor Neutron Flux-High or Inop signal generates a trip signal. This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The APRM System is divided into two groups of channels with three inputs into each trip system. The system is designed to allow one channel in each trip system to be bypassed. Four channels of Average Power Range Monitor-Downscale with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. The Intermediate Range Monitor Neutron Flux-High and Inop Functions are also part of the OPERABILITY of the Average Power Range Monitor-Downscale Function (i.e., if either of these IRM Functions cannot send a signal to the Average Power Range Monitor-Downscale Function, the associated Average Power Range Monitor-Downscale channel is considered inoperable).

The Allowable Value is based upon ensuring that the APRMs are in the linear scale range when transfers are made between APRMs and IRMs.

This Function is required to be OPERABLE in MODE 1 since this is when the APRMs are the primary indicators of reactor power.

2.e. Average Power Range Monitor—Inop

This signal provides assurance that a minimum number of APRMs are OPERABLE. Anytime an APRM mode switch is moved to any position other than "Operate," an APRM module is unplugged, the electronic operating voltage is low, or the APRM has too few LPRM inputs (< 14), an inoperative trip signal will be received by the RPS, unless the APRM is bypassed. Since only one APRM in each trip system may be bypassed, only one APRM in each trip system may be

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

2.e. Average Power Range Monitor—Inop (continued)

inoperable without resulting in an RPS trip signal. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

Four channels of Average Power Range Monitor—Inop with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal.

There is no Allowable Value for this Function.

This Function is required to be OPERABLE in the MODES where the APRM Functions are required.

3. Reactor Vessel Steam Dome Pressure—High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. The Reactor Vessel Steam Dome Pressure—High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 4, reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux—High signal, not the Reactor Vessel Steam Dome Pressure—High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

High reactor pressure signals are initiated from four pressure transmitters that sense reactor pressure. The Reactor Vessel Steam Dome Pressure—High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure—High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3. Reactor Vessel Steam Dome Pressure - High (continued)

ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required to be OPERABLE in MODES 1 and 2 when the RCS is pressurized and the potential for pressure increase exists.

4. Reactor Vessel Water Level - Low, Level 3

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, a reactor scram is initiated at Level 3 to substantially reduce the heat generated in the fuel from fission. The Reactor Vessel Water Level - Low, Level 3 Function is assumed in the analysis of the recirculation line break (Ref. 6). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the Emergency Core Cooling Systems (ECCS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low, Level 3 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

Four channels of Reactor Vessel Water Level - Low, Level 3 Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

The Reactor Vessel Water Level - Low, Level 3 Allowable Value is selected to ensure that (a) during normal operation the steam dryer skirt is not uncovered (this protects available recirculation pump net positive suction head (NPSH) from significant carryunder), and (b) for transients involving loss of all normal feedwater flow, initiation of the low pressure ECCS subsystems at Reactor Vessel Water - Low Low, Level 1 will not be required.

The Function is required in MODES 1 and 2 where considerable energy exists in the RCS resulting in the limiting transients and accidents. ECCS initiations at Reactor

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

4. Reactor Vessel Water Level—Low, Level 3 (continued)

Vessel Water Level—Low Low, Level 2 and Low Low Low, Level 1 provide sufficient protection for level transients in all other MODES.

5. Main Steam Isolation Valve—Closure

MSIV closure results in loss of the main turbine and the condenser as a heat sink for the nuclear steam supply system and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a Main Steam Isolation Valve—Closure signal before the MSIVs are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. However, for the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux—High Function, along with the S/RVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is not assumed in the overpressurization analysis. Additionally, MSIV closure is assumed in the transients analyzed in Reference 7 (e.g., low steam line pressure, manual closure of MSIVs, high steam line flow).

The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

MSIV closure signals are initiated from position switches located on each of the eight MSIVs. Each MSIV has two position switches; one inputs to RPS trip system A while the other inputs to RPS trip system B. Thus, each RPS trip system receives an input from eight Main Steam Isolation Valve—Closure channels, each consisting of one position switch. The logic for the Main Steam Isolation Valve—Closure Function is arranged such that either the inboard or outboard valve on three or more of the main steam lines must close in order for a scram to occur.

The Main Steam Isolation Valve—Closure Allowable Value is specified to ensure that a scram occurs prior to a significant reduction in steam flow, thereby reducing the severity of the subsequent pressure transient.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

5. Main Steam Isolation Valve—Closure (continued)

Sixteen channels of the Main Steam Isolation Valve—Closure Function, with eight channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude the scram from this Function on a valid signal. This Function is only required in MODE 1 since, with the MSIVs open and the heat generation rate high, a pressurization transient can occur if the MSIVs close. In MODE 2, the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection.

6. Drywell Pressure—High

High pressure in the drywell could indicate a break in the RCPB. A reactor scram is initiated to minimize the possibility of fuel damage and to reduce the amount of energy being added to the coolant and the drywell. The Drywell Pressure—High Function is a secondary scram signal to Reactor Vessel Water Level—Low, Level 3 for LOCA events inside the drywell. However, no credit is taken for a scram initiated from this Function for any of the DBAs analyzed in the FSAR. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and indicative of a LOCA inside primary containment.

Four channels of Drywell Pressure—High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required in MODES 1 and 2 where considerable energy exists in the RCS, resulting in the limiting transients and accidents.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

7a, 7b. Scram Discharge Volume Water Level—High

The SDV receives the water displaced by the motion of the CRD pistons during a reactor scram. Should this volume fill to a point where there is insufficient volume to accept the displaced water, control rod insertion would be hindered. Therefore, a reactor scram is initiated while the remaining free volume is still sufficient to accommodate the water from a full core scram. The two types of Scram Discharge Volume Water Level—High Functions are an input to the RPS logic. No credit is taken for a scram initiated from these Functions for any of the design basis accidents or transients analyzed in the FSAR. However, they are retained to ensure the RPS remains OPERABLE.

SDV water level is measured by two diverse methods. The level in each of the two SDVs is measured by two float type level switches and two thermal probes for a total of eight level signals. The outputs of these devices are arranged so that there is a signal from a level switch and a thermal probe to each RPS logic channel. The level measurement instrumentation satisfies the recommendations of Reference 8.

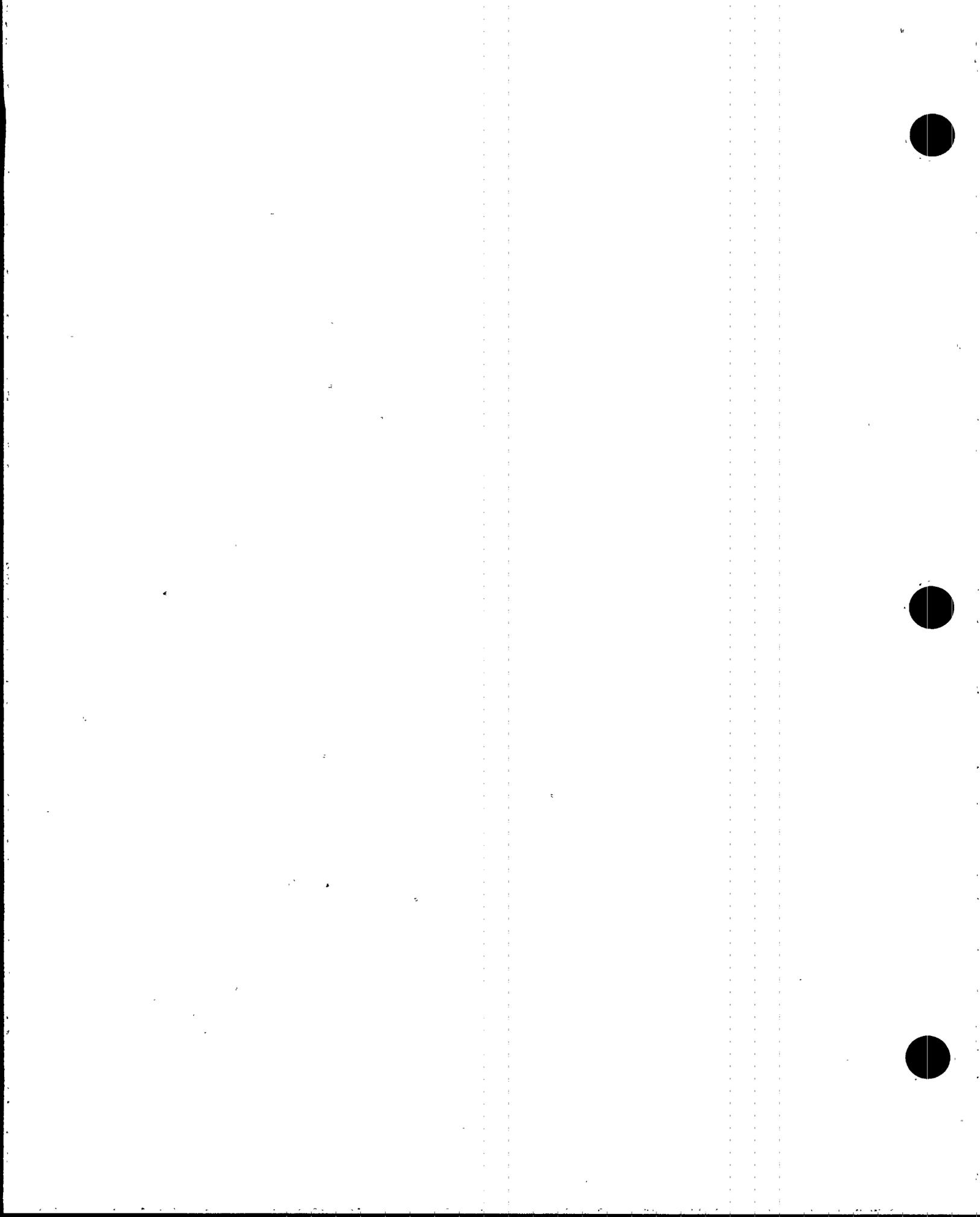
The Allowable Value is chosen low enough to ensure that there is sufficient volume in the SDV to accommodate the water from a full scram.

Four channels of each type of Scram Discharge Volume Water Level—High Function, with two channels of each type in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from these Functions on a valid signal. These Functions are required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn. At all other times, this Function may be bypassed.

8. Turbine Stop Valve—Closure

Closure of the TSVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of

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BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

8. Turbine Stop Valve—Closure
(continued)

the transients that would result from the closure of these valves. The Turbine Stop Valve—Closure Function is the primary scram signal for the turbine trip event analyzed in Reference 7. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded.

Turbine Stop Valve—Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve—Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve—Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER \geq 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function.

The Turbine Stop Valve—Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve—Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if any three TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is \geq 30% RTP. This Function is not required when THERMAL POWER is $<$ 30% RTP since the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor Fixed Neutron Flux—High Functions are adequate to maintain the necessary safety margins.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

9. Turbine Control Valve Fast Closure, Trip Oil
Pressure—Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 7. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure—Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure transmitter is associated with each control valve, and the signal from each transmitter is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER \geq 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function.

The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is \geq 30% RTP. This Function is not required when THERMAL POWER is $<$ 30% RTP, since the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor Fixed Neutron Flux—High Functions are adequate to maintain the necessary safety margins.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

10. Reactor Mode Switch—Shutdown Position

The Reactor Mode Switch—Shutdown Position Function provides signals, via the manual scram logic channels, directly to the scram pilot solenoid power circuits. These manual scram logic channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with four channels, each of which provides input into one of the RPS logic channels.

There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on reactor mode switch position.

Two channels of Reactor Mode Switch—Shutdown Position Function, with one channel in each trip system, are available and required to be OPERABLE. The Reactor Mode Switch—Shutdown Position Function is required to be OPERABLE in MODES 1 and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

11. Manual Scram

The Manual Scram push button channels provide signals, via the manual scram logic channels, directly to the scram pilot solenoid power circuits. These manual scram logic channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

There is one Manual Scram push button channel for each of the two RPS manual scram logic channels. In order to cause a scram it is necessary that each channel in both manual scram trip systems be actuated.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

11. Manual Scram (continued)

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Two channels of Manual Scram with one channel in each manual scram trip system are available and required to be OPERABLE in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

12. RPS Channel Test Switches

There are four RPS Channel Test Switches, one associated with each of the four automatic scram logic channels (A1, A2, B1, and B2). These keylock switches allow the operator to test the OPERABILITY of each individual logic channel without the necessity of using a scram function trip. When the RPS Channel Test Switch is placed in test, the associated scram logic channel is deenergized and OPERABILITY of the channel's scram contactors can be confirmed. The RPS Channel Test Switches are not specifically credited in the accident analysis. However, because the Manual Scram Function at Browns Ferry Nuclear Plant is not configured the same as the generic model in Reference 9, the RPS Channel Test Switches are included in the analysis in Reference 11. Reference 11 concludes that the Surveillance Frequency extensions for RPS functions, described in Reference 9, are not affected by the difference in configuration since each automatic RPS channel has a test switch which is functionally the same as the manual scram switches in the generic model. Weekly testing of scram contactors is credited in Reference 9 with supporting the Surveillance Frequency extension of the RPS functions.

There is no Allowable Value for this Function since the channels are mechanically actuated solely on the position of the switches.

Four channels of the RPS Channel Test Switch Function with two channels in each trip system arranged in a one-out-of-two logic are available and required to be OPERABLE. The function is required in MODES 1 and 2, and in MODE 5 with

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

12. RPS Channel Test Switches (continued)

any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

13. Low Scram Pilot Air Header Pressure

The Low Scram Pilot Air Header Pressure trip performs the same function as the high water level in the scram discharge instrument volume for fast fill events in which the high level instrument response time may not be adequate. A fast fill event is postulated for certain degraded control air events in which the scram outlet valves unseat enough to allow 5 gpm per drive leakage into the scram discharge volume but not enough to cause rod insertion.

The Allowable Value is chosen low enough to ensure that there is sufficient volume in the SDV to accommodate the water from a full scram.

Four channels of Low Scram Pilot Air Header Pressure Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn. At all other times, this Function may be bypassed.

ACTIONS

A Note has been provided to modify the ACTIONS related to RPS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RPS instrumentation channels provide appropriate

(continued)



BASES

ACTIONS
(continued)

compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RPS instrumentation channel.

A.1 and A.2

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (Ref. 9) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function's inoperable channel is in one trip system and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2, and C.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the associated trip system must be placed in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternatively, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a full scram), Condition D must be entered and its Required Action taken.

B.1 and B.2

Condition B exists when, for any one or more Functions, at least one required channel is inoperable in each trip system. In this condition, provided at least one channel per trip system is OPERABLE, the RPS still maintains trip capability for that Function, but cannot accommodate a single failure in either trip system.

Required Actions B.1 and B.2 limit the time the RPS scram logic, for any Function, would not accommodate single failure in both trip systems (e.g., one-out-of-one and one-out-of-one arrangement for a typical four channel Function). The reduced reliability of this logic arrangement was not evaluated in Reference 9 for the 12 hour

(continued)



BASES

ACTIONS

B.1 and B.2 (continued)

Completion Time. Within the 6 hour allowance, the associated Function will have all required channels OPERABLE or in trip (or any combination) in one trip system.

Completing one of these Required Actions restores RPS to a reliability level equivalent to that evaluated in Reference 9, which justified a 12 hour allowable out of service time as presented in Condition A. The trip system in the more degraded state should be placed in trip or, alternatively, all the inoperable channels in that trip system should be placed in trip (e.g., a trip system with two inoperable channels could be in a more degraded state than a trip system with four inoperable channels if the two inoperable channels are in the same Function while the four inoperable channels are all in different Functions). The decision of which trip system is in the more degraded state should be based on prudent judgment and take into account current plant conditions (i.e., what MODE the plant is in). If this action would result in a scram or RPT, it is permissible to place the other trip system or its inoperable channels in trip.

The 6 hour Completion Time is judged acceptable based on the remaining capability to trip, the diversity of the sensors available to provide the trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram.

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram or RPT), Condition D must be entered and its Required Action taken.

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip

(continued)



BASES

ACTIONS

C.1 (continued)

(or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

D.1

Required Action D.1 directs entry into the appropriate Condition referenced in Table 3.3.1.1-1. The applicable Condition specified in the Table is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A, B, or C and the associated Completion Time has expired, Condition D will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1, F.1, and G.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. The allowed Completion Times are reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems. In addition, the Completion Time of Required Action E.1 is consistent with the Completion Time provided in LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)."

H.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in

(continued)



BASES

ACTIONS

H.1 (continued)

trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by immediately initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are, therefore, not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

SURVEILLANCE
REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each RPS instrumentation Function are located in the SRs column of Table 3.3.1.1-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains RPS trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 3) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary.

SR 3.3.1.1.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.1.1 (continued)

something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoints," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate within 2% RTP of calculated MFLPD. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading, between performances of SR 3.3.1.1.7.

A restriction to satisfying this SR when $< 25\%$ RTP is provided that requires the SR to be met only at $\geq 25\%$ RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when $< 25\%$ RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At $\geq 25\%$ RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.1.2 (continued)

per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.3

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

As noted, SR 3.3.1.1.3 is not required to be performed when entering MODE 2 from MODE 1, since testing of the MODE 2 required IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

A Frequency of 7 days provides an acceptable level of system average unavailability over the Frequency interval and is based on reliability analysis (Ref. 9).

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. A Frequency of 7 days provides an acceptable level of system average availability over the Frequency and is based on the reliability analysis of Reference 9. (The RPS Channel Test Switch Function's CHANNEL FUNCTIONAL TEST Frequency was credited in the analysis to extend many automatic scram Functions' Frequencies.)

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.1.1.5 and SR 3.3.1.1.6

These Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power operation for monitoring core reactivity status.

The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a neutron flux region without adequate indication. This is required prior to withdrawing SRMs from the fully inserted position since indication is being transitioned from the SRMs to the IRMs.

The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (by initiating a rod block) if adequate overlap is not maintained. Overlap between IRMs and APRMs exists when sufficient IRMs and APRMs concurrently have onscale readings such that the transition between MODE 1 and MODE 2 can be made without either APRM downscale rod block, or IRM upscale rod block. Overlap between SRMs and IRMs similarly exists when, prior to withdrawing the SRMs from the fully inserted position, IRMs are above mid-scale on range 1 before SRMs have reached the upscale rod block.

As noted, SR 3.3.1.1.6 is only required to be met during entry into MODE 2 from MODE 1. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in MODE 2).

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channels that are required in the current MODE or condition should be declared inoperable.

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.1.1.7

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 effective full power hours Frequency is based on operating experience with LPRM sensitivity changes.

SR 3.3.1.1.8, SR 3.3.1.1.12 and SR 3.3.1.1.16

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 92 day Frequency of SR 3.3.1.1.8 is based on the reliability analysis of Reference 9.

The 184 day Frequency of SR 3.3.1.1.16 for the scram pilot air header low pressure trip function is based on the functional reliability previously demonstrated by this function, the need for minimizing the radiation exposure associated with the functional testing of this function, and the increased risk to plant availability while the plant is in a half-scram condition during the performance of the functional testing versus the limited increase in reliability that would be obtained by the more frequent functional testing.

The 18 month Frequency of SR 3.3.1.1.12 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13 (continued)

adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. For the APRM Simulated Thermal Power-High Function, SR 3.3.1.1.9 also includes calibrating the associated recirculation loop flow channel. For MSIV-Closure, SDV Water Level-High (Float Switch), and TSV-Closure Functions, SR 3.3.1.1.13 also includes physical inspection and actuation of the switches.

Note 1 to SR 3.3.1.1.9 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 effective full power hours LPRM calibration against the TIPS (SR 3.3.1.1.7). A second Note for SR 3.3.1.1.9 is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The Frequency of SR 3.3.1.1.9 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.10 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.1.11

The Average Power Range Monitor Flow Biased Simulated Thermal Power-High Function uses the recirculation loop drive flows to vary the trip setpoint. This SR ensures that the total loop drive flow signals from the flow units used to vary the setpoint are appropriately compared to a

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.1.11 (continued)

The Frequency of 18 months is based on system design considerations which do not support flow unit bypass during operation. Thus, this calibration is performed during refueling outages.

SR 3.3.1.1.14

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods (LCO 3.1.3), and SDV vent and drain valves (LCO 3.1.8), overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.1.1.15

This SR ensures that scrams initiated from the Turbine Stop Valve-Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure-Low Functions will not be inadvertently bypassed when THERMAL POWER is $\geq 30\%$ RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint.

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at $\geq 30\%$ RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve-Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure-Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve-Closure and Turbine Control Valve Fast Closure, Trip

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.1.15 (continued)

Oil Pressure-Low Functions are enabled), this SR is met and the channel is considered OPERABLE.

The Frequency of 18 months is based on engineering judgment and reliability of the components.

REFERENCES

1. FSAR, Section 7.2.
 2. FSAR, Chapter 14.
 3. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.
 4. FSAR, Appendix N.
 5. FSAR, Section 14.6.2.
 6. FSAR, Section 6.5.
 7. FSAR, Section 14.5.
 8. P. Check (NRC) letter to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980.
 9. NEDC-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.
 10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
 11. MED-32-0286, "Technical Specification Improvement Analysis for Browns Ferry Nuclear Plant, Unit 2," October 1995.
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B 3.3 INSTRUMENTATION

B 3.3.1.2 Source Range Monitor (SRM) Instrumentation

BASES

BACKGROUND

The SRMs provide the operator with information relative to the neutron flux level at very low flux levels in the core. As such, the SRM indication is used by the operator to monitor the approach to criticality and determine when criticality is achieved. The SRMs are maintained fully inserted until the count rate is greater than a minimum allowed count rate (a control rod block is set at this condition). After SRM to intermediate range monitor (IRM) overlap is demonstrated (as required by SR 3.3.1.1.5), the SRMs are normally fully withdrawn from the core.

The SRM subsystem of the Neutron Monitoring System (NMS), as described in Reference 1, consists of four channels. Each of the SRM channels can be bypassed, but only one at any given time, by the operation of a bypass switch. Each channel includes one detector that can be physically positioned in the core. Each detector assembly consists of a miniature fission chamber with associated cabling, signal conditioning equipment, and electronics associated with the various SRM functions. The signal conditioning equipment converts the current pulses from the fission chamber to analog DC currents that correspond to the count rate. Each channel also includes indication, alarm, and control rod blocks. However, this LCO specifies OPERABILITY requirements only for the monitoring and indication functions of the SRMs.

During refueling, shutdown, and low power operations, the primary indication of neutron flux levels is provided by the SRMs or special movable detectors connected to the normal SRM circuits. The SRMs provide monitoring of reactivity changes during fuel or control rod movement and give the control room operator early indication of subcritical multiplication that could be indicative of an approach to criticality.

APPLICABLE SAFETY ANALYSES

Prevention and mitigation of prompt reactivity excursions during refueling and low power operation is provided by LCO 3.9.1, "Refueling Equipment Interlocks"; LCO 3.1.1,

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

"SHUTDOWN MARGIN (SDM)"; LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"; IRM Neutron Flux-High and Average Power Range Monitor (APRM) Neutron Flux-High, Setdown Functions; and LCO 3.3.2.1, "Control Rod Block Instrumentation."

The SRMs have no safety function and are not assumed to function during any FSAR design basis accident or transient analysis. However, the SRMs provide the only on scale monitoring of neutron flux levels during startup and refueling. Therefore, they are being retained in Technical Specifications.

LCO

During startup in MODE 2, three of the four SRM channels are required to be OPERABLE to monitor the reactor flux level prior to and during control rod withdrawal, subcritical multiplication and reactor criticality, and neutron flux level and reactor period until the flux level is sufficient to maintain the IRMs on Range 3 or above. All but one of the channels are required in order to provide a representation of the overall core response during those periods when reactivity changes are occurring throughout the core.

In MODES 3 and 4, with the reactor shut down, two SRM channels provide redundant monitoring of flux levels in the core.

In MODE 5, during a spiral offload or reload, an SRM outside the fueled region will no longer be required to be OPERABLE, since it is not capable of monitoring neutron flux in the fueled region of the core. Thus, CORE ALTERATIONS are allowed in a quadrant with no OPERABLE SRM in an adjacent quadrant provided the Table 3.3.1.2-1, footnote (b), requirement that the bundles being spiral reloaded or spiral offloaded are all in a single fueled region containing at least one OPERABLE SRM is met. Spiral reloading and offloading encompass reloading or offloading a cell on the edge of a continuous fueled region (the cell can be reloaded or offloaded in any sequence).

In nonspiral routine operations, two SRMs are required to be OPERABLE to provide redundant monitoring of reactivity

(continued)



BASES

LCO
(continued)

changes occurring in the reactor core. Because of the local nature of reactivity changes during refueling, adequate coverage is provided by requiring one SRM to be OPERABLE in the quadrant of the reactor core where CORE ALTERATIONS are being performed, and the other SRM to be OPERABLE in an adjacent quadrant containing fuel. These requirements ensure that the reactivity of the core will be continuously monitored during CORE ALTERATIONS.

Special movable detectors, according to footnote (c) of Table 3.3.1.2-1, may be used in place of the normal SRM nuclear detectors. These special detectors must be connected to the normal SRM circuits in the NMS, such that the applicable neutron flux indication can be generated. These special detectors provide more flexibility in monitoring reactivity changes during fuel loading, since they can be positioned anywhere within the core during refueling. They must still meet the location requirements of SR 3.3.1.2.2 and all other required SRs for SRMs.

For an SRM channel to be considered OPERABLE, it must be providing neutron flux monitoring indication.

APPLICABILITY

The SRMs are required to be OPERABLE in MODES 2, 3, 4, and 5 prior to the IRMs being on scale on Range 3 to provide for neutron monitoring. In MODE 1, the APRMs provide adequate monitoring of reactivity changes in the core; therefore, the SRMs are not required. In MODE 2, with IRMs on Range 3 or above, the IRMs provide adequate monitoring and the SRMs are not required.

ACTIONS

A.1 and B.1

In MODE 2, with the IRMs on Range 2 or below, SRMs provide the means of monitoring core reactivity and criticality. With any number of the required SRMs inoperable, the ability to monitor neutron flux is degraded. Therefore, a limited time is allowed to restore the inoperable channels to OPERABLE status.

(continued)



BASES

ACTIONS.

A.1 and B.1 (continued)

Provided at least one SRM remains OPERABLE, Required Action A.1 allows 4 hours to restore the required SRMs to OPERABLE status. This time is reasonable because there is adequate capability remaining to monitor the core, there is limited risk of an event during this time, and there is sufficient time to take corrective actions to restore the required SRMs to OPERABLE status or to establish alternate IRM monitoring capability. During this time, control rod withdrawal and power increase is not precluded by this Required Action. Having the ability to monitor the core with at least one SRM, proceeding to IRM Range 3 or greater (with overlap required by SR 3.3.1.1.5), and thereby exiting the Applicability of this LCO, is acceptable for ensuring adequate core monitoring and allowing continued operation.

With three required SRMs inoperable, Required Action B.1 allows no positive changes in reactivity (control rod withdrawal must be immediately suspended) due to inability to monitor the changes. Required Action A.1 still applies and allows 4 hours to restore monitoring capability prior to requiring control rod insertion. This allowance is based on the limited risk of an event during this time, provided that no control rod withdrawals are allowed, and the desire to concentrate efforts on repair, rather than to immediately shut down, with no SRMs OPERABLE.

C.1

In MODE 2, if the required number of SRMs is not restored to OPERABLE status within the allowed Completion Time, the reactor shall be placed in MODE 3. With all control rods fully inserted, the core is in its least reactive state with the most margin to criticality. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 in an orderly manner and without challenging plant systems.

D.1 and D.2

With one or more required SRMs inoperable in MODE 3 or 4, the neutron flux monitoring capability is degraded or nonexistent. The requirement to fully insert all insertable

(continued)



BASES

ACTIONS

D.1 and D.2 (continued)

control rods ensures that the reactor will be at its minimum reactivity level while no neutron monitoring capability is available. Placing the reactor mode switch in the shutdown position prevents subsequent control rod withdrawal by maintaining a control rod block. The allowed Completion Time of 1 hour is sufficient to accomplish the Required Action, and takes into account the low probability of an event requiring the SRM occurring during this interval.

E.1 and E.2

With one or more required SRM inoperable in MODE 5, the ability to detect local reactivity changes in the core during refueling is degraded. CORE ALTERATIONS must be immediately suspended and action must be immediately initiated to insert all insertable control rods in core cells containing one or more fuel assemblies. Suspending CORE ALTERATIONS prevents the two most probable causes of reactivity changes, fuel loading and control rod withdrawal, from occurring. Inserting all insertable control rods ensures that the reactor will be at its minimum reactivity given that fuel is present in the core. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe, conservative position.

Action (once required to be initiated) to insert control rods must continue until all insertable rods in core cells containing one or more fuel assemblies are inserted.

SURVEILLANCE
REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each SRM Applicable MODE or other specified conditions are found in the SRs column of Table 3.3.1.2-1.

SR 3.3.1.2.1 and SR 3.3.1.2.3

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on another channel. It is based on the assumption that instrument channels

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.2.1 and SR 3.3.1.2.3 (continued)

monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency of once every 12 hours for SR 3.3.1.2.1 is based on operating experience that demonstrates channel failure is rare. While in MODES 3 and 4, reactivity changes are not expected; therefore, the 12 hour Frequency is relaxed to 24 hours for SR 3.3.1.2.3. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.2.2

To provide adequate coverage of potential reactivity changes in the core when the fueled region encompasses more than one SRM, one SRM is required to be OPERABLE in the quadrant where CORE ALTERATIONS are being performed, and the other OPERABLE SRM must be in an adjacent quadrant containing fuel. Note 1 states that the SR is required to be met only during CORE ALTERATIONS. It is not required to be met at other times in MODE 5 since core reactivity changes are not occurring. This Surveillance consists of a review of plant logs to ensure that SRMs required to be OPERABLE for given CORE ALTERATIONS are, in fact, OPERABLE. In the event that only one SRM is required to be OPERABLE (when the fueled region encompasses only one SRM), per Table 3.3.1.2-1, footnote (b), only the a. portion of this SR is required. Note 2 clarifies that more than one of the three requirements can be met by the same OPERABLE SRM. The 12 hour Frequency is based upon operating experience and supplements operational controls over refueling activities

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.2.2 (continued)

that include steps to ensure that the SRMs required by the LCO are in the proper quadrant.

SR 3.3.1.2.4

This Surveillance consists of a verification of the SRM instrument readout to ensure that the SRM reading is greater than a specified minimum count rate, which ensures that the detectors are indicating count rates indicative of neutron flux levels within the core. With few fuel assemblies loaded, the SRMs will not have a high enough count rate to satisfy the SR. Therefore, allowances are made for loading sufficient "source" material, in the form of irradiated fuel assemblies, to establish the minimum count rate.

To accomplish this, the SR is modified by Note 1 that states that the count rate is not required to be met on an SRM that has less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies are in the associated core quadrant. With four or less fuel assemblies loaded around each SRM and no other fuel assemblies in the associated core quadrant, even with a control rod withdrawn, the configuration will not be critical. In addition, Note 2 states that this requirement does not have to be met during spiral unloading. If the core is being unloaded in this manner, the various core configurations encountered will not be critical.

The Frequency is based upon channel redundancy and other information available in the control room, and ensures that the required channels are frequently monitored while core reactivity changes are occurring. When no reactivity changes are in progress, the Frequency is relaxed from 12 hours to 24 hours.

SR 3.3.1.2.5 and SR 3.3.1.2.6

Performance of a CHANNEL FUNCTIONAL TEST demonstrates the associated channel will function properly. SR 3.3.1.2.5 is required in MODE 5, and the 7 day Frequency ensures that the channels are OPERABLE while core reactivity changes could be in progress. This Frequency is reasonable, based on

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.2.5 and SR 3.3.1.2.6 (continued)

operating experience and on other Surveillances (such as a CHANNEL CHECK), that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

SR 3.3.1.2.6 is required in MODE 2 with IRMs on Range 2 or below, and in MODES 3 and 4. Since core reactivity changes do not normally take place in MODES 3 and 4 and core reactivity changes are due only to control rod movement in MODE 2, the Frequency has been extended from 7 days to 31 days. The 31 day Frequency is based on operating experience and on other Surveillances (such as CHANNEL CHECK) that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

Verification of the signal to noise ratio also ensures that the detectors are inserted to an acceptable operating level. In a fully withdrawn condition, the detectors are sufficiently removed from the fueled region of the core to essentially eliminate neutrons from reaching the detector. Any count rate obtained while the detectors are fully withdrawn is assumed to be "noise" only.

The Note to the Surveillance allows the Surveillance to be delayed until entry into the specified condition of the Applicability (THERMAL POWER decreased to IRM Range 2 or below). The SR must be performed within 12 hours after IRMs are on Range 2 or below. The allowance to enter the Applicability with the 31 day Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. In this event, the 12 hour Frequency is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances.

SR 3.3.1.2.7

Performance of a CHANNEL CALIBRATION at a Frequency of 92 days verifies the performance of the SRM detectors and

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.2.7 (continued)

associated circuitry. The Frequency considers the plant conditions required to perform the test, the ease of performing the test, and the likelihood of a change in the system or component status. The neutron detectors are excluded from the CHANNEL CALIBRATION (Note 1) because they cannot readily be adjusted. The detectors are fission chambers that are designed to have a relatively constant sensitivity over the range and with an accuracy specified for a fixed useful life.

Note 2 to the Surveillance allows the Surveillance to be delayed until entry into the specified condition of the Applicability. The SR must be performed in MODE 2 within 12 hours of entering MODE 2 with IRMs on Range 2 or below. The allowance to enter the Applicability with the 18 month Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. In this event, the 12 hour Frequency is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances.

REFERENCES

1. FSAR, Section 7.5.4.
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B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch-Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. It is assumed to function to block further control rod withdrawal to preclude a MCPR Safety Limit (SL) violation. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. A signal from one average power range monitor (APRM) channel assigned to each Reactor Protection System (RPS) trip system supplies a reference signal for the RBM channel in the same trip system. If the APRM is indicating less than the low power setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 1).

(continued)



BASES

BACKGROUND
(continued)

The purpose of the RWM is to control rod patterns during startup and shutdown, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence based position indication for each control rod. The RWM also uses feedwater flow and steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 2). The RWM is a single channel system that provides input into both RMCS rod block circuits.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This Function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 3. A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. Note that the RBM setpoint is flow-biased until implementation of ARTS improvements described in Reference 3. However, the generic RWE analysis in Reference 3 is currently applicable to establish required conditions for RBM OPERABILITY.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1. Rod Block Monitor (continued)

The RBM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value to ensure that no single instrument failure can preclude a rod block from this Function. The setpoints are calibrated consistent with applicable setpoint methodology (nominal trip setpoint).

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The RBM is assumed to mitigate the consequences of an RWE event when operating $\geq 29\%$ RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating $< 90\%$ RTP, analyses (Ref. 3) have shown that with an initial MCPR ≥ 1.70 , no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at $\geq 90\%$ RTP with MCPR ≥ 1.40 , no RWE event will result in exceeding the MCPR

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1. Rod Block Monitor (continued)

SL (Ref. 3). Therefore, under these conditions, the RBM is also not required to be OPERABLE.

2. Rod Worth Minimizer

The RWM enforces the banked position withdrawal sequence (BPWS) to ensure that the initial conditions of the CRDA analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References 4, 5, 6, and 7. The BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

The RWM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Since the RWM is designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. 7). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the BPWS. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

Compliance with the BPWS, and therefore OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is < 10% RTP. When THERMAL POWER is > 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Refs. 5 and 7). In MODES 3 and 4, all control rods are required to be inserted into the core; therefore, a CRDA cannot occur. In MODE 5, since only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will be subcritical.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

3. Reactor Mode Switch—Shutdown Position

During MODES 3 and 4, and during MODE 5 when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch—Shutdown Position control rod withdrawal block ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby preserving the assumptions of the safety analysis.

The Reactor Mode Switch—Shutdown Position Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Two channels are required to be OPERABLE to ensure that no single channel failure will preclude a rod block when required. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on reactor mode switch position.

During shutdown conditions (MODE 3, 4, or 5), no positive reactivity insertion events are analyzed because assumptions are that control rod withdrawal blocks are provided to prevent criticality. Therefore, when the reactor mode switch is in the shutdown position, the control rod withdrawal block is required to be OPERABLE. During MODE 5 with the reactor mode switch in the refueling position, the refuel position one-rod-out interlock (LCO 3.9.2, "Refuel Position One-Rod-Out Interlock") provides the required control rod withdrawal blocks.

A.1

With one RBM channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod block function; however, overall reliability is reduced because a single failure in the remaining OPERABLE channel can result in no control rod block capability for the RBM. For this reason, Required Action A.1 requires restoration of the inoperable channel to OPERABLE status. The Completion Time of 24 hours is based on the low probability of an event occurring coincident with a failure in the remaining OPERABLE channel.

(continued)



BASES

ACTIONS
(continued)

B.1

If Required Action A.1 is not met and the associated Completion Time has expired, the inoperable channel must be placed in trip within 1 hour. If both RBM channels are inoperable, the RBM is not capable of performing its intended function; thus, one channel must also be placed in trip. This initiates a control rod withdrawal block, thereby ensuring that the RBM function is met.

The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping of inoperable channels.

C.1, C.2.1.1, C.2.1.2, and C.2.2

With the RWM inoperable during a reactor startup, the operator is still capable of enforcing the prescribed control rod sequence. However, the overall reliability is reduced because a single operator error can result in violating the control rod sequence. Therefore, control rod movement must be immediately suspended except by scram. Alternatively, startup may continue if at least 12 control rods have already been withdrawn, or a reactor startup with an inoperable RWM during withdrawal of one or more of the first 12 rods was not performed in the last 12 months. These requirements minimize the number of reactor startups initiated with the RWM inoperable. Required Actions C.2.1.1 and C.2.1.2 require verification of these conditions by review of plant logs and control room indications. Once Required Action C.2.1.1 or C.2.1.2 is satisfactorily completed, control rod withdrawal may proceed in accordance with the restrictions imposed by Required Action C.2.2. Required Action C.2.2 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff (e.g., a qualified shift technical advisor or reactor engineer).

(continued)



BASES

ACTIONS

C.1, C.2.1.1, C.2.1.2, and C.2.2 (continued)

The RWM may be bypassed under these conditions to allow continued operations. In addition, Required Actions of LCO 3.1.3 and LCO 3.1.6 may require bypassing the RWM, during which time the RWM must be considered inoperable with Condition C entered and its Required Actions taken.

D.1

With the RWM inoperable during a reactor shutdown, the operator is still capable of enforcing the prescribed control rod sequence. Required Action D.1 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff. The RWM may be bypassed under these conditions to allow the reactor shutdown to continue.

E.1 and E.2

With one Reactor Mode Switch-Shutdown Position control rod withdrawal block channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod withdrawal block function. However, since the Required Actions are consistent with the normal action of an OPERABLE Reactor Mode Switch-Shutdown Position Function (i.e., maintaining all control rods inserted), there is no distinction between having one or two channels inoperable.

In both cases (one or both channels inoperable), suspending all control rod withdrawal and initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies will ensure that the core is subcritical with adequate SDM ensured by LCO 3.1.1. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are therefore not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

(continued)



BASES (continued)

SURVEILLANCE
REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by a second Note (Note 2) to indicate that when an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 9) assumption of the average time required to perform a channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control System input.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of 92 days is based on reliability analyses (Ref. 8).

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the RWM is performed by attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs. This test is performed as soon as possible after the applicable conditions are entered. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn at $\leq 10\%$ RTP in MODE 2. As noted, SR 3.3.2.1.3 is not required to be performed until

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.2.1.2 and SR 3.3.2.1.3 (continued)

1 hour after THERMAL POWER is reduced to $\leq 10\%$ RTP in MODE 1. This allows entry into MODE 2 for SR 3.3.2.1.2, and THERMAL POWER reduction to $\leq 10\%$ RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 8).

SR 3.3.2.1.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7.

The Frequency is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.2.1.5

The RWM is automatically bypassed when power is above a specified value. The power level is determined from feedwater flow and steam flow signals. The automatic bypass setpoint must be verified periodically to be $> 10\%$ RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on the trip setpoint methodology utilized for the low power setpoint channel.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.2.1.6

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch-Shutdown Position Function to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch-Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the 18 month Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.2.1.7.

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

REFERENCES

1. FSAR, Section 7.5.8.2.3.
2. FSAR, Section 7.16.5.3.1.k.

(continued)



BASES

REFERENCES
(continued)

3. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3," April 1995.
 4. NEDE-24011-P-A-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, (revision specified in the COLR).
 5. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.
 6. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.
 7. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.
 8. NEDC-30851-P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.
 9. GENE-770-06-1, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.
 10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.3 INSTRUMENTATION

B 3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

BASES

BACKGROUND

The feedwater and main turbine high water level trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow.

With excessive feedwater flow, the water level in the reactor vessel rises toward the high water level reference point, causing the trip of the three feedwater pump turbines and the main turbine.

Reactor Vessel Water Level-High signals are provided by level sensors that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level in the reactor vessel (variable leg). Two channels of Reactor Vessel Water Level-High instrumentation per trip system are provided as input to a two-out-of-two initiation logic that trips the three feedwater pump turbines and the main turbine. There are two trip systems, either of which will initiate a trip. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a main feedwater and turbine trip signal to the trip logic.

A trip of the feedwater pump turbines limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the stop valves protects the turbine from damage due to water entering the turbine.

APPLICABLE
SAFETY ANALYSES

The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The reactor vessel high water level trip indirectly initiates a reactor scram from the main turbine trip (above 30% RTP) and trips the feedwater pumps, thereby terminating

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

the event. The reactor scram mitigates the reduction in MCPR.

Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).

LCO

The LCO requires two channels of the Reactor Vessel Water Level-High instrumentation per trip system to be OPERABLE to ensure that no single instrument failure will prevent the feedwater pump turbines and main turbine trip on a valid Reactor Vessel Water Level-High signal. Both channels in either trip system are needed to provide trip signals in order for the feedwater and main turbine trips to occur. Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.3. The Allowable Value is set to ensure that the thermal limits are not exceeded during the event. The actual setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances,

(continued)



BASES

LCO
(continued) instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

APPLICABILITY The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at $\geq 25\%$ RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," sufficient margin to these limits exists below 25% RTP; therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS A Note has been provided to modify the ACTIONS related to feedwater and main turbine high water level trip instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable feedwater and main turbine high water level trip instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable feedwater and main turbine high water level trip instrumentation channel.

A.1

With one channel inoperable in one trip system, the remaining two OPERABLE channels in the other trip system can provide the required trip signal. However, overall instrumentation reliability is reduced because a single failure in one of the two channels of that trip system

(continued)



BASES

ACTIONS

A.1 (continued)

concurrent with feedwater controller failure, maximum demand event, may result in the instrumentation not being able to perform its intended function. Therefore, continued operation is only allowed for a limited time with one channel inoperable. If the inoperable channel cannot be restored to OPERABLE status within the Completion Time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in a feedwater or main turbine trip), Condition C must be entered and its Required Action taken.

The Completion Time of 7 days is based on the low probability of the event occurring coincident with a single failure in a remaining OPERABLE channel.

B.1

With one or more channels inoperable in each trip system, the feedwater and main turbine high water level trip instrumentation cannot perform its design function (feedwater and main turbine high water level trip capability is not maintained). Therefore, continued operation is only permitted for a 2 hour period, during which feedwater and main turbine high water level trip capability must be restored. The trip capability is considered maintained when sufficient channels are OPERABLE or in trip such that the feedwater and main turbine high water level trip logic will generate a trip signal on a valid signal. This requires that two channels in one trip system be OPERABLE or in trip. If the required channels cannot be restored to OPERABLE status or placed in trip, Condition C must be entered and its Required Action taken.

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of feedwater and main turbine high water level trip instrumentation occurring

(continued)



BASES

ACTIONS

B.1 (continued)

during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1

With the required channels not restored to OPERABLE status or placed in trip, THERMAL POWER must be reduced to < 25% RTP within 4 hours. As discussed in the Applicability section of the Bases, operation below 25% RTP results in sufficient margin to the required limits, and the feedwater and main turbine high water level trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < 25% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 2) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the feedwater pump turbines and main turbine will trip when necessary.

SR 3.3.2.2.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.2.2.1 (continued)

CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels, or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limits.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel status during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.2.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on reliability analysis (Ref. 2).

SR 3.3.2.2.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.2.2.3 (continued)

calibrations consistent with the plant specific setpoint methodology.

The Frequency is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.2.2.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the feedwater and main turbine valves is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a valve is incapable of operating, the associated instrumentation would also be inoperable. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. FSAR, Section 14.5.7.
 2. GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications," February 1991.
 3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.3 INSTRUMENTATION

B 3.3.3.1 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Events. The instruments that monitor these variables are designated as Type A, Category 1, and non-Type A, Category 1, in accordance with Regulatory Guide 1.97 (Ref. 1).

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident. This capability is consistent with the recommendations of Reference 1.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation LCO ensures the OPERABILITY of Regulatory Guide 1.97, Type A variables so that the control room operating staff can:

- Perform the diagnosis specified in the Emergency Operating Instructions (EOIs). These variables are restricted to preplanned actions for the primary success path of Design Basis Accidents (DBAs), (e.g., loss of coolant accident (LOCA)), and
- Take the specified, preplanned, manually controlled actions for which no automatic control is provided, which are required for safety systems to accomplish their safety function.

The PAM instrumentation LCO also ensures OPERABILITY of Category 1, non-Type A, variables so that the control room operating staff can:

- Determine whether systems important to safety are performing their intended functions;

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- Determine the potential for causing a gross breach of the barriers to radioactivity release;
- Determine whether a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and for an estimate of the magnitude of any impending threat.

The plant specific Regulatory Guide 1.97 Analysis (Ref. 2) documents the process that identified Type A and Category 1, non-Type A, variables.

Accident monitoring instrumentation that satisfies the definition of Type A in Regulatory Guide 1.97 meets Criterion 3 of the NRC Policy Statement (Ref. 6). Category 1, non-Type A, instrumentation is retained in Technical Specifications (TS) because they are intended to assist operators in minimizing the consequences of accidents. Therefore, these Category 1 variables are important for reducing public risk.

LCO

LCO 3.3.3.1 requires two OPERABLE channels for all but one Function to ensure that no single failure prevents the operators from being presented with the information necessary to determine the status of the plant and to bring the plant to, and maintain it in, a safe condition following that accident. Furthermore, provision of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

The exception to the two channel requirement is primary containment isolation valve (PCIV) position. In this case, the important information is the status of the primary containment penetrations. The LCO requires one position indicator for each active (e.g., automatic) PCIV. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of passive valve or via system boundary status. If a normally active PCIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position

(continued)



BASES

LCO
(continued)

indication for closed and deactivated valves is not required to be OPERABLE.

The following list is a discussion of the specified instrument Functions listed in Table 3.3.3.1-1.

1. Reactor Steam Dome Pressure

Reactor steam dome pressure is a Category 1 variable provided to support monitoring of Reactor Coolant System (RCS) integrity and to verify operation of the Emergency Core Cooling Systems (ECCS). Two independent pressure transmitters with a range of 0 psig to 1200 psig monitor pressure. Wide range indicators are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

2. Reactor Vessel Water Level

Reactor vessel water level is a Category 1 variable provided to support monitoring of core cooling and to verify operation of the ECCS. Two different range water level channels (Emergency Systems and Post-accident Flood Range) provide the PAM Reactor Vessel Water Level Functions. The water level channels measure from 1/3 of the core height to 221 inches above the top of the active fuel. Water level is measured by two independent differential pressure transmitters for each required channel. The output from these channels is indicated on two independent indicators, which is the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

The reactor vessel water level instruments are not compensated for variation in reactor water density. Function 2.a is calibrated to be most accurate at operational pressure and temperature while Function 2.b is calibrated to be most accurate for accident conditions.

(continued)



BASES

LCO
(continued)

3. Suppression Pool Water Level

Suppression pool water level is a Category 1 variable provided to detect a breach in the reactor coolant pressure boundary (RCPB). This variable is also used to verify and provide long term surveillance of ECCS function. The wide range suppression pool water level measurement provides the operator with sufficient information to assess the status of both the RCPB and the water supply to the ECCS. The wide range water level indicators monitor the suppression pool water level from two feet from the bottom of the pool to five feet above normal water level. Two wide range suppression pool water level signals are transmitted from separate differential pressure transmitters and are continuously recorded and displayed on one recorder and one indicator in the control room. The recorder and indicator are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

4. Drywell Pressure

Drywell pressure is a Category 1 variable provided to detect breach of the RCPB and to verify ECCS functions that operate to maintain RCS integrity. Two different ranges of drywell pressure channels (normal and wide range) receive signals that are transmitted from separate pressure transmitters and are continuously recorded and displayed on two control room recorders and two control room indicators. These recorders and indicators are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

5. Primary Containment Area Radiation (High Range)

Primary containment area radiation (high range) is provided to monitor the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. Two high range primary containment area radiation signals are transmitted from separate radiation detectors and are continuously recorded and displayed on two control room recorders. These recorders are the primary indication used

(continued)



BASES

LCO

5. Primary Containment Area Radiation (High Range)
(continued)

by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

6. Primary Containment Isolation Valve (PCIV) Position

PCIV position is provided for verification of containment integrity. In the case of PCIV position, the important information is the isolation status of the containment penetration. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active PCIV in a containment penetration flow path, i.e., two total channels of PCIV position indication for a penetration flow path with two active valves. For containment penetrations with only one active PCIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration via indicated status of the active valve, as applicable, and prior knowledge of passive valve or system boundary status. If a penetration flow path is isolated, position indication for the PCIV(s) in the associated penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration flow path is not required to be OPERABLE.

The indication for each PCIV consists of green and red indicator lights that illuminate to indicate whether the PCIV is fully open, fully closed, or in a mid-position. Therefore, the PAM specification deals specifically with this portion of the instrument channel.

7. Drywell and Torus Hydrogen Analyzers

Drywell and torus hydrogen analyzers are Category 1 instruments provided to detect high hydrogen or oxygen concentration conditions that represent a potential for containment breach. The drywell and torus hydrogen concentration recorders allow the operators to detect trends in hydrogen concentration in sufficient time to initiate

(continued)



BASES

LCO

7. Drywell and Torus Hydrogen Analyzers (continued)

containment atmospheric dilution if containment atmosphere approaches combustible limits. Hydrogen concentration indication is also important in verifying the adequacy of mitigating actions. High hydrogen concentration is measured by two independent analyzers and continuously recorded and displayed on one control room recorder and one control room indicator. The analyzers have the capability for sampling both the drywell and the torus. These indicators are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

8. Suppression Pool Water Temperature

Suppression pool water temperature is a Category 1 variable provided to detect a condition that could potentially lead to containment breach and to verify the effectiveness of ECCS actions taken to prevent containment breach. The suppression pool water temperature instrumentation allows operators to detect trends in suppression pool water temperature in sufficient time to take action to prevent steam quenching vibrations in the suppression pool. Sixteen temperature sensors are arranged in two groups of two independent and redundant channels, located such that they are sufficient to provide a reasonable measure of bulk pool temperature. The outputs for the sensors are recorded on two independent recorders in the control room. These recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channels.

9. Drywell Atmosphere Temperature

Drywell atmosphere temperature is a Category 1 variable provided to detect a condition that could potentially lead to containment breach and to verify the effectiveness of ECCS actions taken to prevent containment breach. Two wide range drywell atmosphere temperature signals are transmitted from separate temperature transmitters and are continuously recorded and displayed on one control room recorder and one control room indicator. The recorder and indicator are the primary indications used by the operator during an accident.

(continued)



BASES

LCO

9. Drywell Atmosphere Temperature (continued)

Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1 and 2. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1 and 2. In MODES 3, 4, and 5, plant conditions are such that the likelihood of an event that would require PAM instrumentation is extremely low; therefore, PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

Note 1 has been added to the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to diagnose an accident using alternative instruments and methods, and the low probability of an event requiring these instruments.

Notes 2 and 3 have been provided to modify the ACTIONS related to PAM instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable PAM instrumentation channels provide appropriate compensatory measures for separate Functions. As such, Note 2 has been provided to allow separate Condition entry for each inoperable PAM Function. Note 3 has been provided for Function 6 to allow separate Condition entry for each penetration flow path.

(continued)



BASES

ACTIONS
(continued)

A.1

When one or more Functions have one required channel that is inoperable, the required inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channels (or, in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

If a channel has not been restored to OPERABLE status in 30 days, this Required Action specifies initiation of action in accordance with Specification 5.6.6, which requires a written report to be submitted to the NRC. This report discusses the alternate method of monitoring, the results of the root cause evaluation of the inoperability, and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement, since alternative actions are identified before loss of functional capability, and given the likelihood of plant conditions that would require information provided by this instrumentation.

C.1

When one or more Functions have two required channels that are inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the

(continued)



BASES

ACTIONS

C.1 (continued)

PAM Function will be in a degraded condition should an accident occur. Condition C is modified by a Note that excludes hydrogen monitor channels. Condition D provides appropriate Required Actions for two inoperable hydrogen monitor channels.

D.1

When two hydrogen monitor channels are inoperable, one hydrogen monitor channel must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time is based on the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit; and the length of time after the event that operator action would be required to prevent hydrogen accumulation from exceeding this limit.

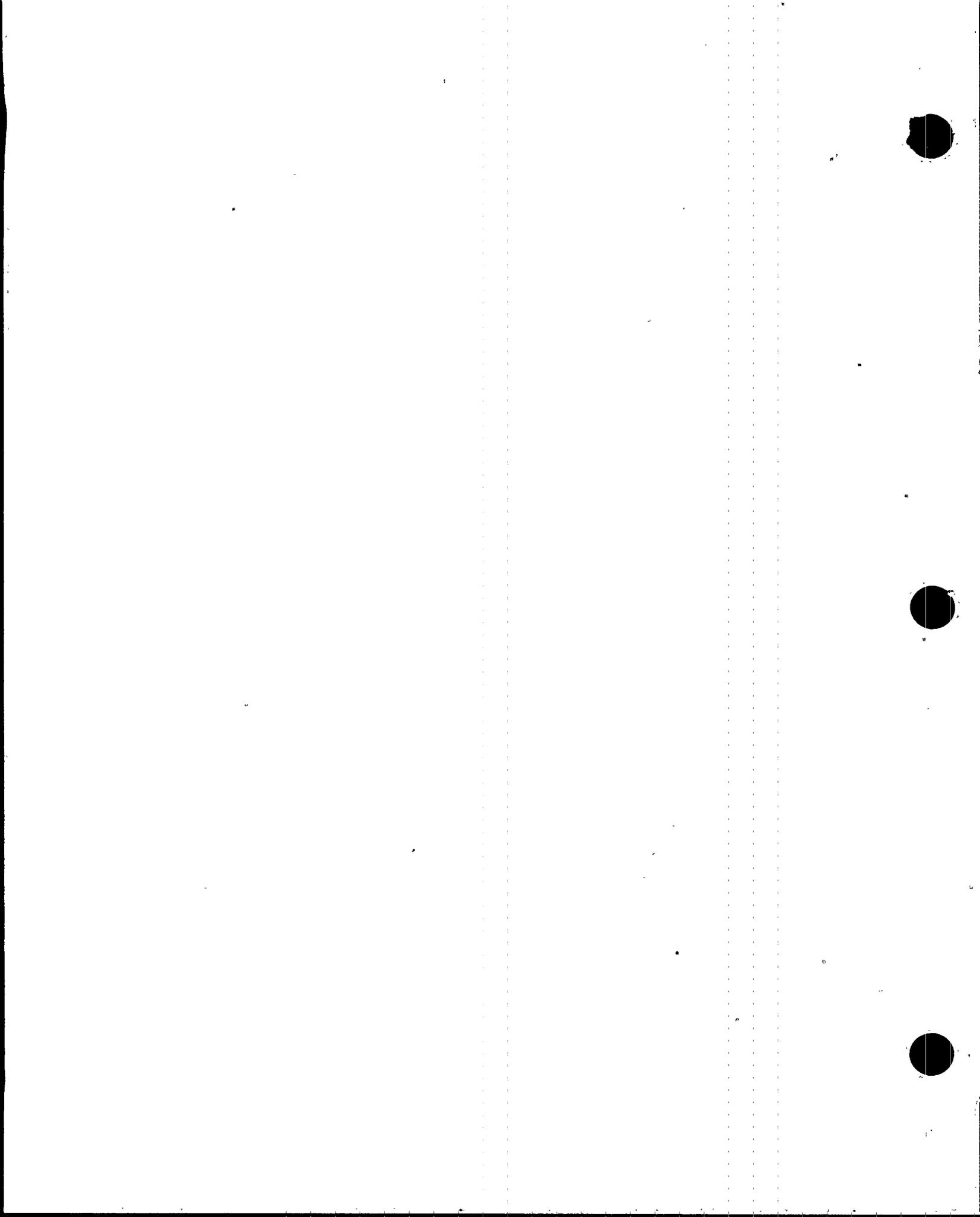
E.1

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.3.1-1. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met any Required Action of Condition C or D, as applicable, and the associated Completion Time has expired, Condition E is entered for that channel and provides for transfer to the appropriate subsequent Condition.

F.1

For the majority of Functions in Table 3.3.3.1-1, if any Required Action and associated Completion Time of Condition C or D are not met, the plant must be brought to a MODE in which the LCO not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)



BASES

ACTIONS
(continued)

G.1

Since alternate means of monitoring primary containment area radiation have been developed and tested, the Required Action is not to shut down the plant, but rather to follow the directions of Specification 5.6.6. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

SURVEILLANCE
REQUIREMENTS

SR 3.3.3.1.1

Performance of the CHANNEL CHECK for each required PAM instrumentation channel once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel against a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrument channels should be compared to each other or to other containment radiation monitoring instrumentation.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency of 31 days is based upon plant operating experience, with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.3.1.1 (continued)

a given Function in any 31 day interval is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of those displays associated with the required channels of this LCO.

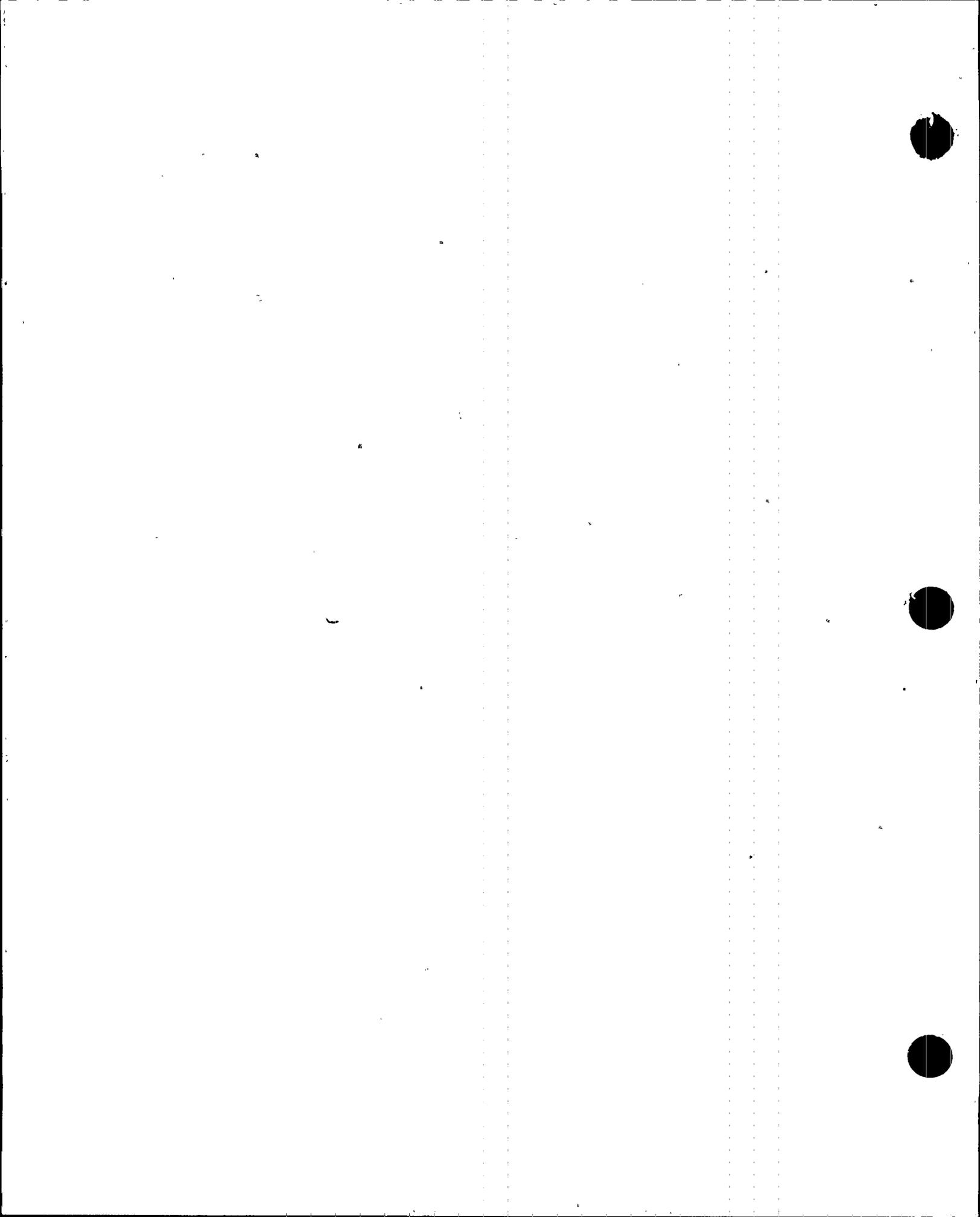
SR 3.3.3.1.2 and SR 3.3.3.1.3

A CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies the channel responds to measured parameter with the necessary range and accuracy. For the PCIV position function, the CHANNEL CALIBRATION consists of verifying the remote indications conform to actual valve positions.

The 92 day Frequency for CHANNEL CALIBRATION of the Drywell and Torus Hydrogen Analyzer is based on operating experience and vendor recommendations. The 18 month Frequency for CHANNEL CALIBRATION of all other PAM instrumentation in Table 3.3.3.1-1 is based on operating experience and consistency with BFN refueling cycles.

REFERENCES

1. Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3, May 1983.
2. TVA Letter from L. M. Mills to H. R. Denton (NRC) dated April 30, 1984.
3. NRC Letter from S.C. Black to S. A. White (TVA), NRC Regulatory Guide 1.97 SER letter, dated June 23, 1988.
4. TVA General Design Criteria No. BFN-50-7307, Revision 4, "Post-Accident Monitoring," dated June 22, 1993.
5. NRC Letter from Joseph F. Williams to Oliver D. Kingsley, Jr., "Regulatory Guide 1.97 - Boiling Water Reactor Neutron Flux Monitoring For the Browns Ferry Nuclear Plant, Units 1, 2, and 3," dated May 3, 1994.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.



B 3.3 INSTRUMENTATION

B 3.3.3.2 Backup Control System

BASES

BACKGROUND

The Backup Control System provides the control room operator with sufficient instrumentation and controls to place and maintain the plant in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility of the control room becoming inaccessible. A safe shutdown condition is defined as MODE 3. With the plant in MODE 3, the Reactor Core Isolation Cooling (RCIC) System, the safety/relief valves, and the Residual Heat Removal System can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the RCIC and the ability to operate the RHR System for decay heat removal from outside the control room allow extended operation in MODE 3.

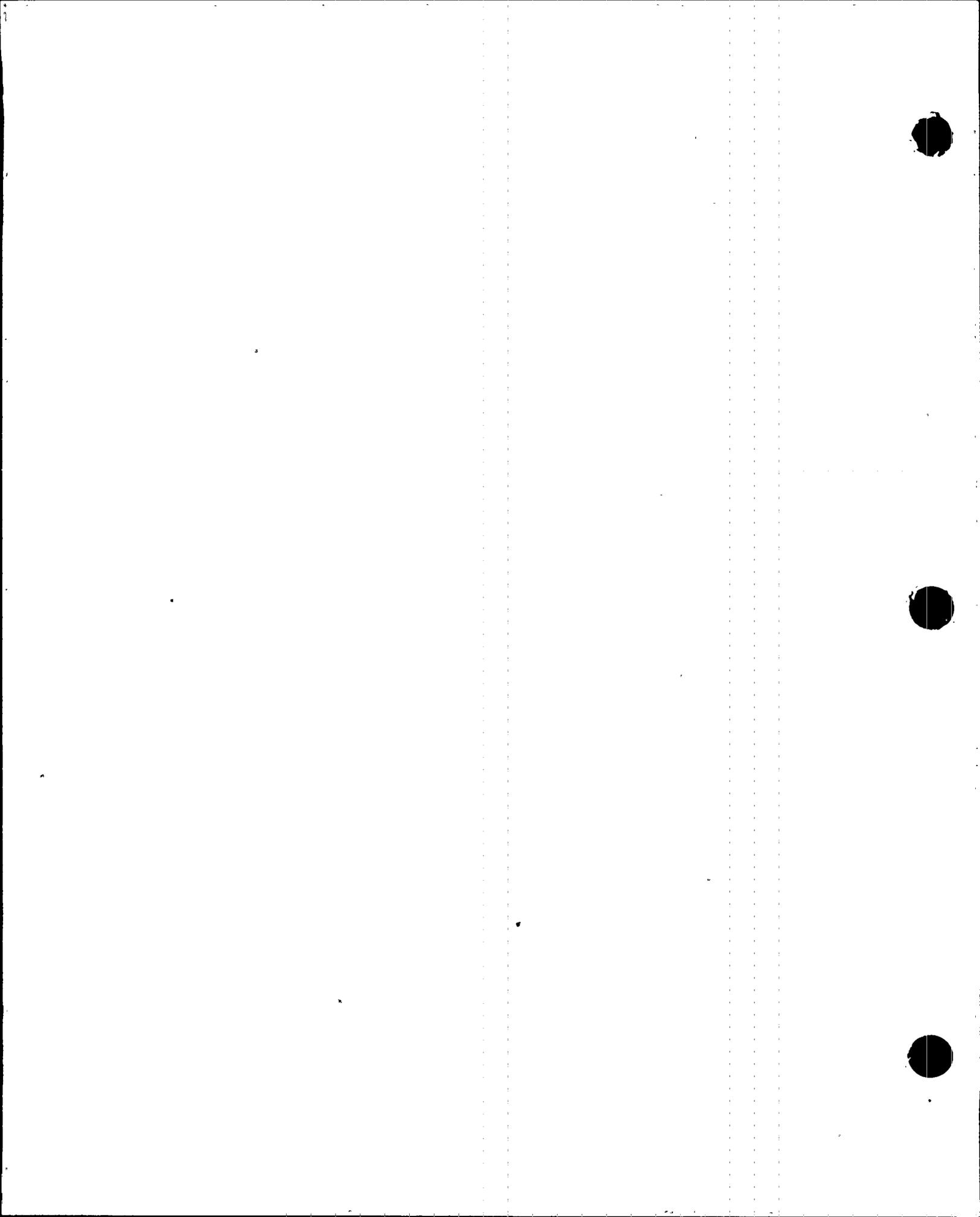
In the event that the control room becomes inaccessible, the operators can establish control at the backup control panel and place and maintain the plant in MODE 3. Not all controls and necessary transfer switches are located at the backup control panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The plant automatically reaches MODE 3 following a plant shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the Backup Control System control and instrumentation Functions ensures that there is sufficient information available on selected plant parameters to place and maintain the plant in MODE 3 should the control room become inaccessible.

APPLICABLE SAFETY ANALYSES

The Backup Control System is required to provide equipment at appropriate locations outside the control room with a design capability to promptly shut down the reactor to MODE 3, including the necessary instrumentation and controls, to maintain the plant in a safe condition in MODE 3.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The criteria governing the design and the specific system requirements of the Backup Control System are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1) and Reference 2.

The Backup Control System is considered an important contributor to reducing the risk of accidents; as such, it meets Criterion 4 of the NRC Policy Statement (Ref. 3).

LCO

The Backup Control System LCO provides the requirements for the OPERABILITY of the instrumentation and controls necessary to place and maintain the plant in MODE 3 from a location other than the control room. The instrumentation and controls typically required are listed in Table B 3.3.3.2-1.

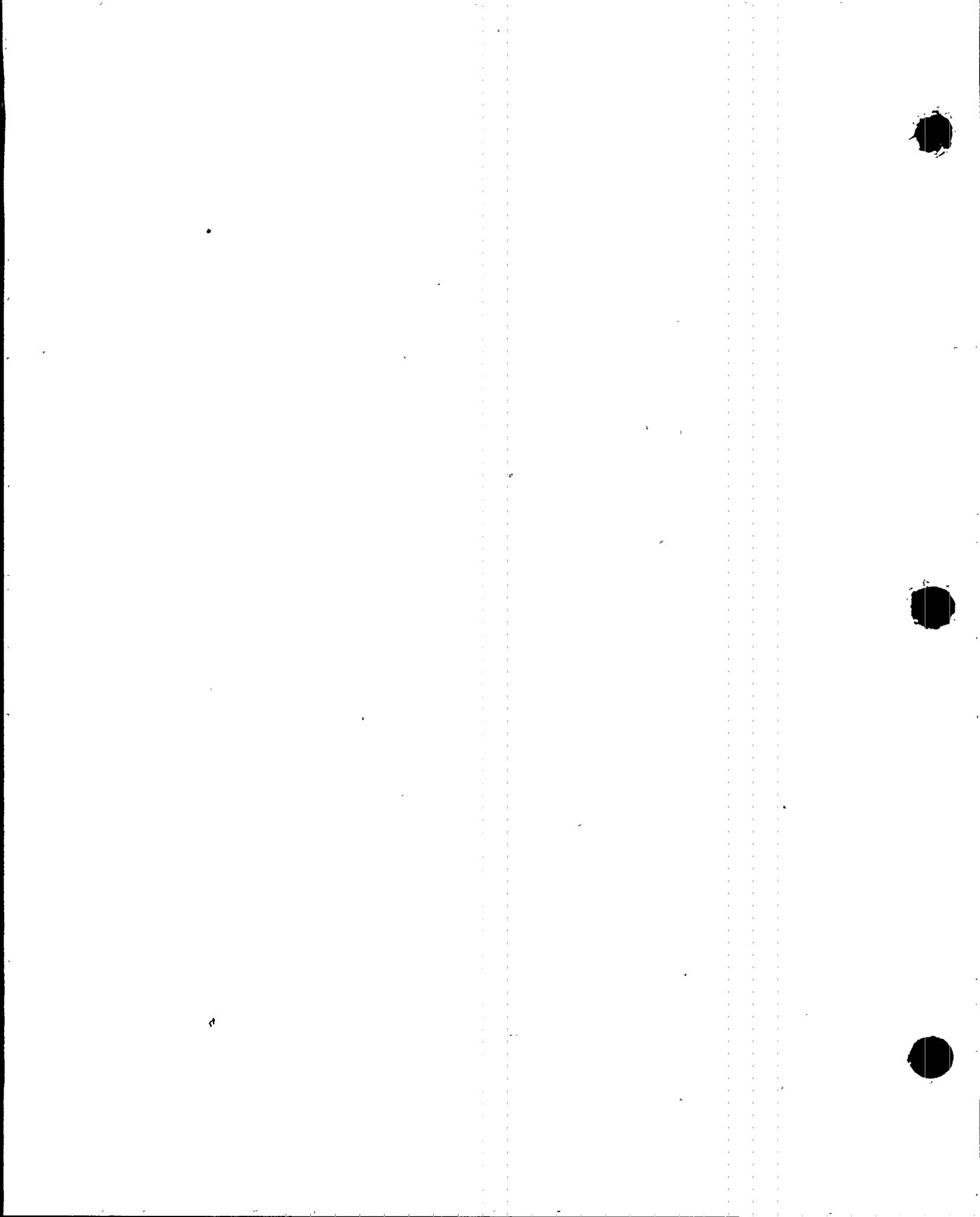
The controls, instrumentation, and transfer switches are those required for:

- Reactor pressure vessel (RPV) pressure control;
- Decay heat removal;
- RPV inventory control; and
- Safety support systems for the above functions, including Residual Heat Removal (RHR) Service Water, Emergency Equipment Cooling Water, and onsite power, including the diesel generators.

The Backup Control System is OPERABLE if all instrument and control channels needed to support the backup control function are OPERABLE. In some cases, Table B 3.3.3.2-1 may indicate that the required information or control capability is available from several alternate sources. In these cases, the Backup Control System is OPERABLE as long as one channel of any of the alternate information or control sources for each Function is OPERABLE.

The Backup Control System instruments and control circuits covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure that the instruments and control circuits will be OPERABLE if plant conditions require that the Backup Control System be placed in operation.

(continued)



BASES (continued)

APPLICABILITY The Backup Control System LCO is applicable in MODES 1 and 2. This is required so that the plant can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODES 3, 4, and 5. In these MODES, the plant is already subcritical and in a condition of reduced Reactor Coolant System energy. Under these conditions, considerable time is available to restore necessary instrument control Functions if control room instruments or control becomes unavailable. Consequently, the TS do not require OPERABILITY in MODES 3, 4, and 5.

ACTIONS A Note is included that excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a plant shutdown. This exception is acceptable due to the low probability of an event requiring this system.

Note 2 has been provided to modify the ACTIONS related to Backup Control System Functions. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable Backup Control System Functions provide appropriate compensatory measures for separate Functions. As such, a Note has been provided that allows separate Condition entry for each inoperable Backup Control System Function.

A.1

Condition A addresses the situation where one or more required Functions of the Backup Control System is inoperable. This includes any Function listed in Table B 3.3.3.2-1, as well as the control and transfer switches.

(continued)



BASES

ACTIONS

A.1 (continued)

The Required Action is to restore the Function to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

B.1

If the Required Action and associated Completion Time of Condition A are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.3.3.2.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. As specified in the Surveillance, a CHANNEL CHECK is only required for those channels that are normally energized.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.3.2.1 (continued)

The Frequency is based upon plant operating experience that demonstrates channel failure is rare.

SR 3.3.3.2.2

SR 3.3.3.2.2 verifies each required Backup Control System transfer switch and control circuit performs the intended function. This verification is performed from the backup control panel and locally, as appropriate. Operation of the equipment from the backup control panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the plant can be placed and maintained in MODE 3 from the backup control panel and the local control stations. Operating experience demonstrates that Backup Control System control channels usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.3.2.3 and SR 3.3.3.2.4

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies the channel responds to measured parameter values with the necessary range and accuracy.

The Frequency of SR 3.3.3.2.3 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The 18 month Frequency of SR 3.3.3.2.4 is based upon operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.
 2. FSAR Section 7.18.
 3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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Table B 3.3.3.2-1 (Page 1 of 4)
Backup Control System Instrumentation and Controls

FUNCTION	REQUIRED NUMBER OF CHANNELS
<u>Instrument Parameter</u>	
1. Reactor Water Level Indication	1
2. Reactor Pressure Indication	1
3. Suppression Pool Temperature Indication	1
4. Suppression Pool Level Indication	1
5. Drywell Pressure Indication	1
6. Drywell Temperature Indication	1
7. EECW Flow Indication	2 (1/Header)
8. RCIC Flow Indication	1
9. RCIC Turbine Speed Indication	1
10. RCIC Turbine Trip Alarm	1
11. RCIC Turbine Bearing Oil High Temperature Alarm	1
<u>Transfer/Control Parameter</u>	
12. MSRV Transfer & Control	3 (1/MSRV)
13. MSIV Transfer & Control (Closure Only)	8 (1/MSIV)
14. Main Steam Drain Line Isolation Valves	2 (1/valve)
15. RHRSW Pumps	12 (1/pump)
16. RHRSW Discharge Valves for RHR Loop I Heat Exchangers	2 (1/valve)
17. RCW Pumps 1D and 3D (Trip Function Only)	2 (1/pump)



Table B 3.3.3.2-1 (Page 2 of 4)
Backup Control System Instrumentation and Controls

FUNCTION	REQUIRED NUMBER OF CHANNELS
<u>Transfer/Control Parameter (continued)</u>	
18. 4-kV Fire Pumps A, B, and C	3 (1/pump)
19. Recirculation System Sample Line Isolation Valves	2 (1/valve)
20. EECW Sectionalizing Valves	8 (1/valve)
21. RHRSW to EECW Motor-Operated Crosstie Valves	2 (1/valve)
22. EECW Supply to RBCCW Heat Exchangers	6 (1/valve)
23. Recirculation Pump Discharge Valve (RHR Loop I LPCI)	1
24. RWCU Drain to Main Condenser Hotwell Isolation Valve	1
25. RWCU Drain to Radwaste Isolation Valve	1
26. RBCCW Pump Controls	2 (1/pump)
27. Drywell Cooler RBCCW Flow Control Valves	10 (1/cooler)
28. Drywell Cooler Fan Controls	10 (1/cooler)
29. RHR Shutdown Cooling Inboard Containment Isolation Valve	1
30. RHR Shutdown Cooling Outboard Containment Isolation Valve	1
31. RCIC Steam Supply Isolation Valves	2 (1/valve)
32. RCIC Steam Pot Drain Line Steam Trap Bypass	1



Table B 3.3.3.2-1 (Page 3 of 4)
Backup Control System Instrumentation and Controls

FUNCTION	REQUIRED NUMBER OF CHANNELS
<u>Transfer/Control Parameter (continued)</u>	
33. RCIC Steam Pot Drain to Main Condenser Isolation	1 (1 switch for 2 valves)
34. RCIC Drain to Radwaste Isolation	1 (1 switch for 2 valves)
35. RCIC Turbine Steam Supply Valve	1
36. RCIC Turbine Stop Valve	1
37. RCIC Pump Suction From Suppression Pool	2 (1/valve)
38. RCIC Pump Suction From Condensate Storage Tank	1
39. RCIC Lube Oil Cooler Cooling Water Supply	1
40. RCIC Pump Minimum Flow Bypass	1
41. RCIC Pump Discharge	1
42. RCIC Test Return to Condensate Storage Tank	1
43. RCIC Injection Valve to Reactor Vessel	1
44. RCIC Barometric Condenser Condensate Pump	1
45. RCIC Barometric Condenser Vacuum Pump	1
46. HPCI Turbine Steam Supply Valve (Isolation Function Only)	1
47. RHR Pump Controls	4 (1/pump)
48. RHR Loop I Motor Operated Valves	17 (1/valve)



Table B 3.3.3.2-1 (Page 4 of 4)
Backup Control System Instrumentation and Controls

FUNCTION	REQUIRED NUMBER OF CHANNELS
<u>Transfer/Control Parameter (continued)</u>	
49. Core Spray Pumps (Trip & Lock-out Function Only)	4 (1/pump)
50. CRD Pump 1B	1
51. CRD Pump Discharge Valves	2 (1/valve)
52. Scram Discharge Volume Isolation Pilot Valve	1



B 3.3 INSTRUMENTATION

B 3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

BASES

BACKGROUND

The EOC-RPT instrumentation initiates a recirculation pump trip (RPT) to reduce the peak reactor pressure and power resulting from turbine trip or generator load rejection transients to provide additional margin to core thermal M CPR Safety Limits (SLs).

The need for the additional negative reactivity in excess of that normally inserted on a scram reflects end of cycle reactivity considerations. Flux shapes at the end of cycle are such that the control rods may not be able to ensure that thermal limits are maintained by inserting sufficient negative reactivity during the first few feet of rod travel upon a scram caused by Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure-Low or Turbine Stop Valve (TSV)-Closure. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity at a faster rate than the control rods can add negative reactivity.

The EOC-RPT instrumentation, as shown in Reference 1, is composed of sensors that detect initiation of closure of the TSVs or fast closure of the TCVs, combined with relays, logic circuits, and fast acting circuit breakers that interrupt power from the recirculation pump motor generator (MG) set generators to each of the recirculation pump motors. The channels include electronic equipment (e.g., trip relays) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an EOC-RPT signal to the trip logic. When the RPT breakers trip open, the recirculation pumps coast down under their own inertia. The EOC-RPT has two identical trip systems, either of which can actuate an RPT.

Each EOC-RPT trip system is a two-out-of-two logic for each Function; thus, either two TSV-Closure or two TCV Fast Closure, Trip Oil Pressure-Low signals are required for a trip system to actuate. If either trip system actuates, both recirculation pumps will trip. There are two EOC-RPT breakers in series per recirculation pump. One trip system trips one of the two EOC-RPT breakers for each recirculation

(continued)



BASES

BACKGROUND
(continued)

pump, and the second trip system trips the other EOC-RPT breaker for each recirculation pump.

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The TSV-Closure and the TCV Fast Closure, Trip Oil Pressure-Low Functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the increase in neutron flux, heat flux, and reactor pressure, and to increase the margin to the MCPR SL. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection are summarized in References 2, 3, and 4.

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps after initiation of closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than a scram alone, resulting in an increased margin to the MCPR SL. Alternatively, MCPR limits for an inoperable EOC-RPT, as specified in the COLR, are sufficient to prevent violation of the MCPR Safety Limit. The EOC-RPT function is automatically disabled when turbine first stage pressure is $< 30\%$ RTP.

EOC-RPT instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

The OPERABILITY of the EOC-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.1.3. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). Channel OPERABILITY also includes the associated EOC-RPT breakers. Each channel (including the associated EOC-RPT breakers) must also respond within its assumed response time.

Allowable Values are specified for each EOC-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The nominal setpoints are selected to ensure that the setpoints-do not exceed the Allowable Value

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Each Allowable Value specified is more conservative than the analytical limit assumed in the transient and accident analysis in order to account for instrument uncertainties appropriate to the Function. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., TSV position), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip relay) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The specific Applicable Safety Analysis, LCO, and Applicability discussions are listed below on a Function by Function basis.

Alternatively, since this instrumentation protects against a MCPR SL violation, with the instrumentation inoperable, modifications to the MCPR limits (LCO 3.2.2) may be applied to allow this LCO to be met. The MCPR penalty for the EOC-RPT inoperable condition is specified in the COLR.

Turbine Stop Valve Closure

Closure of the TSVs and a main turbine trip result in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TSV-Closure in anticipation of the transients that would result from closure of these valves. EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

Turbine Stop Valve - Closure (continued)

Closure of the TSVs is determined by measuring the position of each valve. There are two separate position switches associated with each stop valve, the signal from each switch being assigned to a separate trip channel. The logic for the TSV-Closure Function is such that two or more TSVs must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER \geq 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function. Four channels of TSV-Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT, from this Function on a valid signal. The TSV-Closure Allowable Value is selected to detect imminent TSV closure.

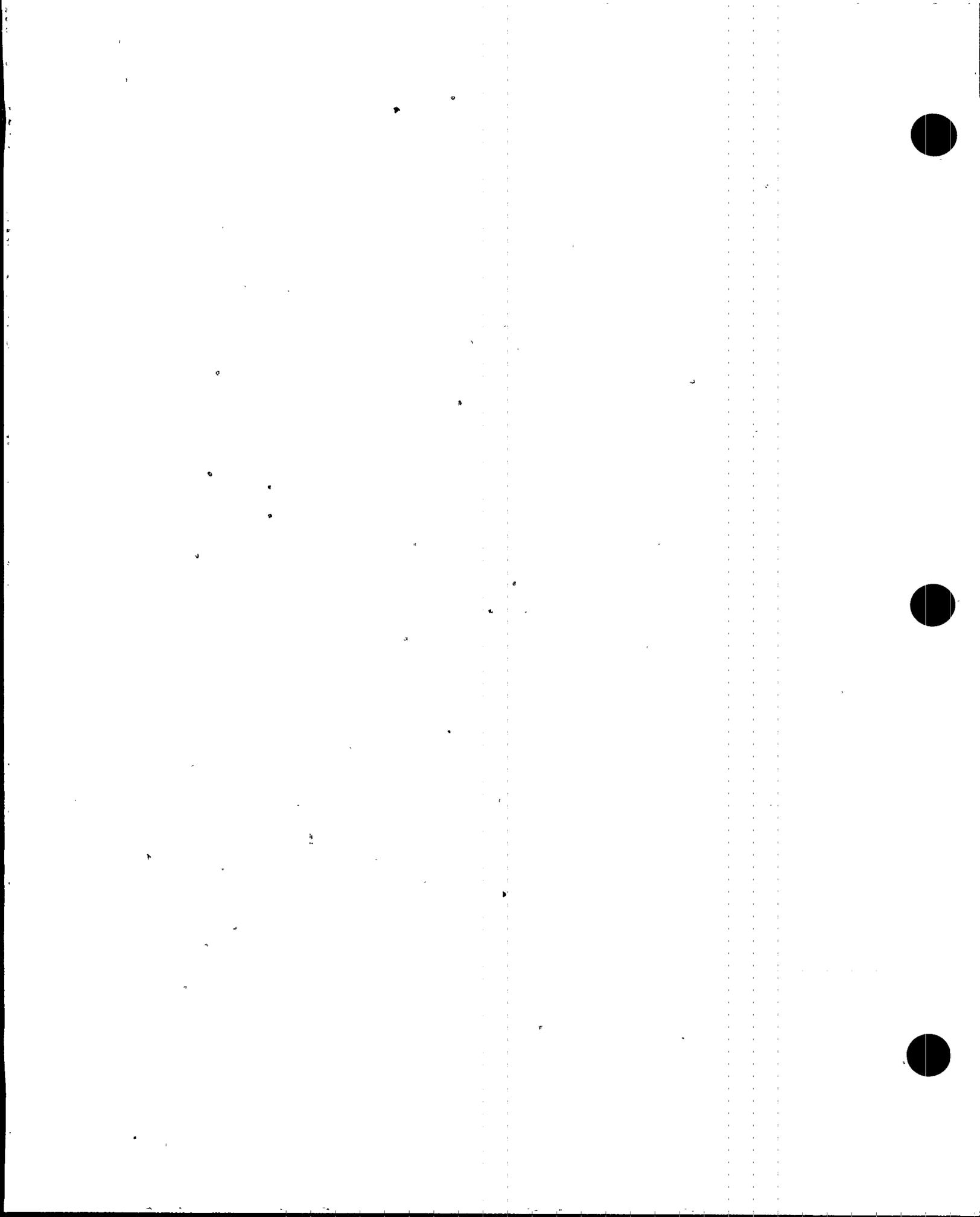
This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is \geq 30% RTP. Below 30% RTP, the Reactor Vessel Steam Dome Pressure-High and the Average Power Range Monitor (APRM) Fixed Neutron Flux-High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary margin to the MCPR Safety Limit.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure-Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Fast closure of the TCVs is determined by measuring the electrohydraulic control fluid pressure at each control valve. There is one pressure transmitter associated with each control valve, and the signal from each transmitter is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure-Low Function is such that two or more TCVs must be closed (pressure transmitter trips)

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

Turbine Control Valve Fast Closure, Trip Oil Pressure- Low
(continued)

to produce an EOC-RPT. This Function must be enabled at THERMAL POWER \geq 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function. Four channels of TCV Fast Closure, Trip Oil Pressure-Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure-Low Allowable Value is selected high enough to detect imminent TCV fast closure.

This protection is required consistent with the safety analysis whenever THERMAL POWER is \geq 30% RTP. Below 30% RTP, the Reactor Vessel Steam Dome Pressure-High and the APRM Fixed Neutron Flux-High Functions of the RPS are adequate to maintain the necessary safety margins.

ACTIONS

A Note has been provided to modify the ACTIONS related to EOC-RPT instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable EOC-RPT instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable EOC-RPT instrumentation channel.

A.1

With one or more channels inoperable, but with EOC-RPT trip capability maintained (refer to Required Actions B.1 and B.2 Bases), the EOC-RPT System is capable of performing the intended function. However, the reliability and redundancy

(continued)



BASES

ACTIONS

A.1 (continued)

of the EOC-RPT instrumentation is reduced such that a single failure in the remaining trip system could result in the inability of the EOC-RPT System to perform the intended function. Therefore, only a limited time is allowed to restore compliance with the LCO. Because of the diversity of sensors available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of an EOC-RPT, 72 hours is provided to restore the inoperable channels (Required Action A.1) or apply the EOC-RPT inoperable MCPR limit. Alternately, the inoperable channels may be placed in trip (Required Action A.2) since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable breaker, since this may not adequately compensate for the inoperable breaker (e.g., the breaker may be inoperable such that it will not open). If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an RPT, or if the inoperable channel is the result of an inoperable breaker), Condition C must be entered and its Required Actions taken.

B.1 and B.2

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining EOC-RPT trip capability. A Function is considered to be maintaining EOC-RPT trip capability when sufficient channels are OPERABLE or in trip, such that the EOC-RPT System will generate a trip signal from the given Function on a valid signal and both recirculation pumps can be tripped. Alternately, Required Action B.2 requires the MCPR limit for inoperable EOC-RPT, as specified in the COLR, to be applied. This also restores the margin to MCPR assumed in the safety analysis.

The 2 hour Completion Time is sufficient time for the operator to take corrective action, and takes into account

(continued)



BASES

ACTIONS

B.1 and B.2 (continued)

the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1

With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to < 30% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER to < 30% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 5) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the recirculation pumps will trip when necessary.

SR 3.3.4.1.1

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on reliability analysis of Reference 5.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.4.1.2

This SR ensures that an EOC-RPT initiated from the TSV-Closure and TCV Fast Closure, Trip Oil Pressure-Low Functions will not be inadvertently bypassed when THERMAL POWER is $\geq 30\%$ RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at $\geq 30\%$ RTP, either due to open main turbine bypass valves or other reasons), the affected TSV-Closure and TCV Fast Closure, Trip Oil Pressure-Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met with the channel considered OPERABLE.

The Frequency of 18 months is based on engineering judgment and reliability of the components.

SR 3.3.4.1.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.4.1.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the pump breakers is included as a part of this test, overlapping the LOGIC SYSTEM FUNCTIONAL TEST, to provide complete testing of the associated safety function. Therefore, if a breaker is incapable of operating, the associated instrument channel(s) would also be inoperable.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.4.1.4 (continued)

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. FSAR, Figure 7.9-2 (EOC-RPT logic diagram).
 2. FSAR, Section 7.9.4.5.
 3. FSAR, Sections 14.5.1.1 and 14.5.1.3.
 4. FSAR, Section 4.3.5.
 5. GENE-770-06-1, "Bases For Changes To Surveillance Test Intervals And Allowed Out-Of-Service Times For Selected Instrumentation Technical Specifications," February 1991.
 6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.3 INSTRUMENTATION

B 3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

BASES

BACKGROUND

The ATWS-RPT System initiates an RPT, adding negative reactivity, following events in which a scram does not (but should) occur, to lessen the effects of an ATWS event. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases. When Reactor Vessel Water Level-Low or Reactor Steam Dome Pressure-High setpoint is reached, the recirculation pump motor breakers trip.

The ATWS-RPT System (Ref. 1) includes sensors, relays, bypass capability, circuit breakers, and switches that are necessary to cause initiation of an RPT. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an ATWS-RPT signal to the trip logic.

The ATWS-RPT consists of two independent trip systems, with two channels of Reactor Steam Dome Pressure-High and two channels of Reactor Vessel Water Level-Low in each trip system. Each ATWS-RPT trip system is a two-out-of-two logic for each function. Thus, either two Reactor Water Level-Low or two Reactor Pressure-High signals are needed to trip a trip system. The outputs of the channels in a trip system are combined in a logic so that either trip system will trip both recirculation pumps (by tripping the respective motor breakers).

There are two motor breakers provided for each of the two recirculation pumps for a total of four breakers. The output of each trip system is provided to both recirculation pump breakers.

(continued)



BASES (continued)

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The ATWS-RPT is not assumed in the safety analysis. The ATWS-RPT initiates an RPT to aid in preserving the integrity of the fuel cladding following events in which a scram does not, but should, occur. Based on its contribution to the reduction of overall plant risk, however, the instrumentation meets Criterion 4 of the NRC Policy Statement (Ref. 3).

The OPERABILITY of the ATWS-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.2.3. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). ATWS-RPT Channel OPERABILITY also includes the associated recirculation pump motor breakers. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Allowable Values are specified for each ATWS-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and environmental effects are accounted for.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

The individual Functions are required to be OPERABLE in MODE 1 to protect against catastrophic/multiple failures of the Reactor Protection System by providing a diverse trip to mitigate the consequences of a postulated ATWS event. The Reactor Steam Dome Pressure-High and Reactor Vessel Water Level-Low Functions are required to be OPERABLE in MODE 1, since the reactor is producing significant power and the recirculation system could be at high flow. During this MODE, the potential exists for pressure increases or low water level, assuming an ATWS event. In MODE 2, the reactor is at low power and the recirculation system is at low flow; thus, the potential is low for a pressure increase or low water level, assuming an ATWS event. Therefore, the ATWS-RPT is not necessary. In MODES 3 and 4, the reactor is shut down with all control rods inserted; thus, an ATWS event is not significant and the possibility of a significant pressure increase or low water level is negligible. In MODE 5, the one rod out interlock ensures that the reactor remains subcritical; thus, an ATWS event is not significant. In addition, the reactor pressure vessel (RPV) head is not fully tensioned and no pressure transient threat to the reactor coolant pressure boundary (RCPB) exists.

The specific Applicable Safety Analyses and LCO discussions are listed below on a Function by Function basis.

a. Reactor Vessel Water Level - Low

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the ATWS-RPT System is initiated at Level 2 to aid in maintaining level above the top of the active fuel. The reduction of core flow reduces the neutron flux and THERMAL POWER and, therefore, the rate of coolant boiloff.

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

a. Reactor Vessel Water Level - Low
(continued)

Four channels of Reactor Vessel Water Level - Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Vessel Water Level - Low Allowable Value is chosen so that the system will not be initiated after a Level 3 scram with feedwater still available, and for convenience with the reactor core isolation cooling initiation.

b. Reactor Steam Dome Pressure - High

Excessively high RPV pressure may rupture the RCPB. An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This increases neutron flux and THERMAL POWER, which could potentially result in fuel failure and overpressurization. The Reactor Steam Dome Pressure - High Function initiates an RPT for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation. For the overpressurization event, the RPT aids in the termination of the ATWS event and, along with the safety/relief valves, limits the peak RPV pressure to less than the ASME Section III Code limits.

The Reactor Steam Dome Pressure - High signals are initiated from four pressure transmitters that monitor reactor steam dome pressure. Four channels of Reactor Steam Dome Pressure - High, with two channels in each trip system, are available and are required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Steam Dome Pressure - High Allowable Value is chosen to provide an adequate margin to the ASME Section III Code limits.

ACTIONS

A Note has been provided to modify the ACTIONS related to ATWS-RPT instrumentation channels. Section 1.3, Completion

(continued)



BASES

ACTIONS
(continued)

Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable ATWS-RPT instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable ATWS-RPT instrumentation channel.

A.1 and A.2

With one or more channels inoperable, but with ATWS-RPT capability for each Function maintained (refer to Required Actions B.1 and C.1 Bases), the ATWS-RPT System is capable of performing the intended function. However, the reliability and redundancy of the ATWS-RPT instrumentation is reduced, such that a single failure in the remaining trip system could result in the inability of the ATWS-RPT System to perform the intended function. Therefore, only a limited time is allowed to restore the inoperable channels to OPERABLE status. Because of the diversity of sensors available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of ATWS-RPT, 14 days is provided to restore the inoperable channel (Required Action A.1). Alternately, the inoperable channel may be placed in trip (Required Action A.2), since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable breaker, since this may not adequately compensate for the inoperable breaker (e.g., the breaker may be inoperable such that it will not open). If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel would result in an RPT), or if the inoperable channel is the result of an inoperable breaker, Condition D must be entered and its Required Actions taken.

(continued)



BASES

ACTIONS
(continued)

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining ATWS-RPT trip capability. A Function is considered to be maintaining ATWS-RPT trip capability when sufficient channels are OPERABLE or in trip such that the ATWS-RPT System will generate a trip signal from the given Function on a valid signal, and both recirculation pumps can be tripped. This requires one channel of the Function in each trip system to be OPERABLE or in trip, and the recirculation pump motor breakers to be OPERABLE or in trip.

The 72 hour Completion Time is sufficient for the operator to take corrective action (e.g., restoration or tripping of channels) and takes into account the likelihood of an event requiring actuation of the ATWS-RPT instrumentation during this period and that one Function is still maintaining ATWS-RPT trip capability.

C.1

Required Action C.1 is intended to ensure that appropriate Actions are taken if multiple, inoperable, untripped channels within both Functions result in both Functions not maintaining ATWS-RPT trip capability. The description of a Function maintaining ATWS-RPT trip capability is discussed in the Bases for Required Action B.1 above.

The 1 hour Completion Time is sufficient for the operator to take corrective action and takes into account the likelihood of an event requiring actuation of the ATWS-RPT instrumentation during this period.

D.1

With any Required Action and associated Completion Time not met, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, both to reach MODE 2 from full power conditions and to remove a

(continued)



BASES

ACTIONS

D.1 (continued)

recirculation pump from service in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into the associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 2) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the recirculation pumps will trip when necessary.

SR 3.3.4.2.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.4.2.1 (continued)

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.4.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 2.

SR 3.3.4.2.3

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.4.2.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the pump breakers is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a breaker is incapable of operating, the associated instrument channel(s) would be inoperable.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.4.2.4 (continued)

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. FSAR Section 7.19.
 2. GENE-770-06-1, "Bases for Changes To Surveillance Test Intervals and Allowed Out-of-Service Times For Selected Instrumentation Technical Specifications," February 1991.
 3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.3 INSTRUMENTATION

B 3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

BASES

BACKGROUND

The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient.

For most anticipated operational occurrences and Design Basis Accidents (DBAs), a wide range of dependent and independent parameters are monitored.

The ECCS instrumentation actuates core spray (CS), low pressure coolant injection (LPCI), high pressure coolant injection (HPCI), Automatic Depressurization System (ADS), and the diesel generators (DGs). The equipment involved with each of these systems is described in the Bases for LCO 3.5.1, "ECCS-Operating."

Core Spray System

The CS System may be initiated by automatic means. Each pump can be controlled manually by a control room remote switch. Automatic initiation occurs for conditions of Reactor Vessel Water Level-Low Low Low, Level 1 or both Drywell Pressure-High and Reactor Steam Dome Pressure-Low. Reactor water level and drywell pressure are monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of these trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic (i.e., two trip systems) for each Function. The Reactor Steam Dome Pressure-Low variable is monitored by two transmitters for each subsystem. The outputs from these transmitters are connected to relays arranged in a one-out-of-two logic.

The high drywell pressure initiation signal is a sealed in signal and must be manually reset. Upon receipt of an initiation signal, if normal AC power is available, the four core spray pumps start one at a time, in order, at 0, 7, 14, and 21 seconds. If normal AC power is not available,

(continued)



BASES

BACKGROUND

Core Spray System (continued)

the four core spray pumps start seven seconds after standby power becomes available. (The LPCI pumps start as soon as standby power is available.)

The CS test line isolation valve is closed on a CS initiation signal to allow full system flow assumed in the accident analyses.

The CS pump discharge flow is monitored by a flow switch. When the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow assumed in the accident analysis.

The CS System logic also receives signals from transmitters which monitor the pressure in the reactor to ensure that, before the injection valves open, the reactor pressure has fallen to a value below the CS System's maximum design pressure. Reactor pressure is monitored by four redundant transmitters, which are, in turn, connected to four trip units (two per subsystem). The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two logic for each CS subsystem.

Low Pressure Coolant Injection System

The LPCI is an operating mode of the Residual Heat Removal (RHR) System, with two LPCI subsystems. The LPCI subsystems may be initiated by automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level-Low Low Low, Level 1 or both Drywell Pressure-High and Reactor Steam Dome Pressure-Low. Each of these diverse variables is monitored by four redundant transmitters, which, in turn, are connected to four trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic (i.e., two trip systems) for each Function.

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BASES

BACKGROUND

Low Pressure Coolant Injection System (continued)

Once an initiation signal is received by the LPCI control circuitry, the signal is sealed in until manually reset.

Upon receipt of an initiation signal, if normal AC power is available, the four RHR (LPCI) pumps start one at a time, in order; at 0, 7, 14, and 21 seconds. If normal AC power is not available, the four pumps start simultaneously, with no delay, as soon as the standby power source is available.

Each LPCI subsystem's discharge flow is monitored by a flow switch. When a pump is running and discharge flow is low enough so that pump overheating may occur, the respective minimum flow return line valve is opened. If flow is above the minimum flow setpoint, the valve is automatically closed. However, LPCI flow rates assumed in the LOCA analyses can be achieved with the minimum flow valve in the open position.

The RHR test line suppression pool cooling isolation valve, suppression pool spray isolation valves, and containment spray isolation valves (which are also PCIVs) are also closed on a LPCI initiation signal to allow the full system flow assumed in the accident analyses and maintain primary containment isolated in the event LPCI is not operating.

The LPCI System monitors the pressure in the reactor to ensure that, before an injection valve opens, the reactor pressure has fallen to a value below the LPCI System's maximum design pressure. The variable is monitored by four redundant transmitters, which are, in turn, connected to multiple trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic. Additionally, these instruments function to initiate closure of the recirculation pump discharge valves to ensure that LPCI flow does not bypass the core when it injects into the recirculation lines.

Low reactor water level in the shroud is detected by two additional instruments which inhibit the manual initiation of other modes of RHR (e.g., suppression pool cooling) when

(continued)



BASES

BACKGROUND

Low Pressure Coolant Injection System (continued)

LPCI is required. Manual overrides for the inhibit logic are provided.

High Pressure Coolant Injection System

The HPCI System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level—Low Low, Level 2 or Drywell Pressure—High. Each of these variables is monitored by four redundant transmitters, which are, in turn, connected to multiple trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each Function.

The HPCI pump discharge flow is monitored by a flow switch. Upon automatic initiation, when the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow assumed in the accident analysis.

The HPCI test line isolation valve is closed upon receipt of a HPCI initiation signal to allow the full system flow assumed in the accident analysis.

The HPCI System also monitors the water levels in the HPCI pump supply header from the condensate storage tank (CST) and the suppression pool because these are the two sources of water for HPCI operation. Reactor grade water in the CST is the normal source. Upon receipt of a HPCI initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position) unless both suppression pool suction valves are open. If the water level in the HPCI pump supply header from the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. Two level switches are used to detect low water level in the HPCI pump supply header from the CST. Either switch can cause the suppression pool suction valves to open and the CST suction valve to close.

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BASES

BACKGROUND

High Pressure Coolant Injection System (continued)

The suppression pool suction valves also automatically open and the CST suction valve closes if high water level is detected in the suppression pool. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

The HPCI provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level-High, Level 8 trip, at which time the HPCI turbine trips, which causes the turbine's stop valve to close. The logic is two-out-of-two to provide high reliability of the HPCI System. The HPCI System automatically restarts if a Reactor Vessel Water Level-Low Low, Level 2 signal is subsequently received.

Automatic Depressurization System

The ADS may be initiated by either automatic or manual means. Automatic initiation occurs when signals indicating Reactor Vessel Water Level-Low Low Low, Level 1; Drywell Pressure-High or ADS High Drywell Pressure Bypass Timer; confirmed Reactor Vessel Water Level-Low, Level 3; and CS or LPCI Pump Discharge Pressure-High are all present and the ADS Initiation Timer has timed out. There are two transmitters each for Reactor Vessel Water Level-Low Low Low, Level 1 and Drywell Pressure-High, and one transmitter for confirmed Reactor Vessel Water Level-Low, Level 3 in each of the two ADS trip systems. Each of these transmitters connects to a trip unit, which then drives a relay whose contacts form the initiation logic.

Each ADS trip system includes a time delay between satisfying the initiation logic and the actuation of the ADS valves. The ADS Initiation Timer time delay setpoint chosen is long enough that the HPCI has sufficient operating time to recover to a level above Level 1, yet not so long that the LPCI and CS Systems are unable to adequately cool the fuel if the HPCI fails to maintain that level. An alarm in the control room is annunciated when either of the ADS Initiation Timers is timing. Resetting the ADS initiation signals resets the ADS Initiation Timers.

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BASES

BACKGROUND

Automatic Depressurization System (continued)

The ADS also monitors the discharge pressures of the four LPCI pumps and the four CS pumps. Each ADS trip system includes two discharge pressure permissive switches from two of the four CS pumps (A and B for one trip system and C and D for the other trip system) and one discharge pressure permissive switch for each LPCI pump. The signals are used as a permissive for ADS actuation, indicating that there is a source of core coolant available once the ADS has depressurized the vessel. CS pumps (A or B and either C or D) or any one of the four LPCI pumps is sufficient to permit automatic depressurization.

The ADS logic in each trip system is arranged in two strings. Each string has a contact from each of the following variables: Reactor Vessel Water Level—Low Low Low, Level 1; Drywell Pressure—High; or Low Water Level Actuation Timer. One of the two strings in each trip system must also have a confirmed Reactor Vessel Water Level—Low, Level 3. All contacts in both logic strings must close, the ADS initiation timer must time out, and a CS or LPCI pump discharge pressure signal must be present to initiate an ADS trip system. Either the A or B trip system will cause all the ADS relief valves to open. Once the Drywell Pressure—High signal, the ADS High Drywell Pressure Bypass Timer, or the ADS initiation signal is present, it is individually sealed in until manually reset.

Manual inhibit switches are provided in the control room for the ADS; however, their function is not required for ADS OPERABILITY (provided ADS is not inhibited when required to be OPERABLE).

Diesel Generators

The DGs may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level—Low Low Low, Level 1 or both Drywell Pressure—High and Reactor Steam Dome Pressure—Low. The DGs are also initiated upon loss of voltage signals. (Refer to the Bases for LCO 3.3.8.1, "Loss of Power (LOP) Instrumentation," for a discussion of these signals.) Each of these diverse variables is monitored by four redundant transmitters, which are, in turn, connected to four trip

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BASES

BACKGROUND

Diesel Generators (continued)

units. The outputs of the four trip units are connected to relays whose contacts are connected to a one-out-of-two taken twice logic to initiate all eight DGs (A, B, C, D, 3A, 3B, 3C, and 3D). The DGs receive their initiation signals from the CS System initiation logic. The DGs can also be started manually from the control room and locally from the associated DG room. The DG initiation signal is a sealed in signal and must be manually reset. The DG initiation logic is reset by resetting the associated ECCS initiation logic. Upon receipt of a loss of coolant accident (LOCA) initiation signal, each DG is automatically started, is ready to load in approximately 10 seconds, and will run in standby conditions (rated voltage and speed, with the DG output breaker open). The DGs will only energize their respective Engineered Safety Feature buses if a loss of offsite power occurs. (Refer to Bases for LCO 3.3.8.1.)

Emergency Equipment Cooling Water (EECW) System

The EECW System, which distributes cooling water supplied by the RHR Service Water System pumps that are assigned as the principal supply to the EECW System (RHRSW pumps A3, B3, C3 and D3), may be initiated by automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level—Low Low Low, Level 1 or Drywell Pressure—High with a Reactor Steam Dome Pressure—Low permissive. Each of these diverse variables is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The EECW System receives its initiation signals from the DG initiation logic and the CS System initiation logic. The two RHRSW pumps (B3 and D3) assigned to EECW and powered from shutdown boards in Units 1 and 2 will start automatically in less than 32.5 seconds after starting of a diesel generator or 30 seconds for a core spray pump in Units 1 and 2. The two RHRSW pumps (A3 and C3) assigned to EECW and powered from shutdown boards in Unit 3 will start automatically in less than 32.5 seconds after starting of a diesel generator or 30 seconds for a core spray pump in Unit 3. In addition, the signals that start the A3 and C3 pumps and the B3 and D3 pumps also start the B1 and D1 pumps and the A1 and C1 pumps, respectively, when they are valved into the EECW header.

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BASES (continued)

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The actions of the ECCS are explicitly assumed in the safety analyses of References 1, 2, and 3. The ECCS is initiated to preserve the integrity of the fuel cladding by limiting the post LOCA peak cladding temperature to less than the 10 CFR 50.46 limits.

ECCS instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 5). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the ECCS instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.5.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). Each ECCS subsystem must also respond within its assumed response time. Table 3.3.5.1-1, footnote (b), is added to show that certain ECCS instrumentation Functions are also required to be OPERABLE to perform DG initiation and actuation of other Technical Specifications (TS) equipment.

Allowable Values are specified for each ECCS Function specified in the table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined, accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived

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BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions that may require ECCS (or DG) initiation to mitigate the consequences of a design basis transient or accident. To ensure reliable ECCS and DG function, a combination of Functions is required to provide primary and secondary initiation signals.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Core Spray and Low Pressure Coolant Injection Systems

1.a. 2.a. Reactor Vessel Water Level—Low Low Low, Level 1

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The low pressure ECCS, associated DGs, and EECW System are initiated at Level 1 to ensure that core spray and flooding functions are available to prevent or minimize fuel damage. The Reactor Vessel Water Level—Low Low Low, Level 1 is one of the Functions assumed to be OPERABLE and capable of initiating the ECCS during the transients analyzed in References 1 and 3. In addition, the Reactor Vessel Water Level—Low Low Low, Level 1 Function is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level—Low Low Low, Level 1 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

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BASES

**APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY**

1.a, 2.a. Reactor Vessel Water Level—Low Low Low, Level 1
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The Reactor Vessel Water Level—Low Low Low, Level 1 Allowable Value is chosen to allow time for the low pressure injection/spray subsystems to activate and provide adequate cooling.

Four channels of Reactor Vessel Water Level—Low Low Low, Level 1 Function are only required to be OPERABLE when the ECCS, DG(s), or EECW System are required to be OPERABLE to ensure that no single instrument failure can preclude ECCS, DG, and EECW initiation. Refer to LCO 3.5.1 and LCO 3.5.2, "ECCS—Shutdown," for Applicability Bases for the low pressure ECCS subsystems; LCO 3.7.2, "Emergency Equipment Cooling (EECW) Systems and Ultimate Heat Sink (UHS)," for Applicability Bases for EECW System; and LCO 3.8.1, "AC Sources—Operating"; and LCO 3.8.2, "AC Sources—Shutdown," for Applicability Bases for the DGs.

1.b, 2.b. Drywell Pressure—High

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS, associated DGs, and EECW System are initiated upon receipt of the Drywell Pressure—High Function in order to minimize the possibility of fuel damage. The Drywell Pressure—High Function, along with the Reactor Pressure—Low Function, is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

The Drywell Pressure—High Function is required to be OPERABLE when the ECCS, DG, or EECW System are required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE. Thus, four channels of the CS and LPCI Drywell Pressure—High Function are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude ECCS, DG, and

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BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1.b. 2.b. Drywell Pressure— High (continued)

EECW System initiation. In MODES 4 and 5, the Drywell Pressure—High Function is not required, since there is insufficient energy in the reactor to pressurize the primary containment to Drywell Pressure—High setpoint. Refer to LCO 3.5.1 for Applicability Bases for the low pressure ECCS subsystems, LCO 3.7.2 for Applicability Bases for the EECW System, and to LCO 3.8.1 for Applicability Bases for the DGs.

1.c. 2.c. Reactor Steam Dome Pressure— Low (Injection Permissive and ECCS Initiation)

Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. This ensures that, prior to opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure. The Reactor Steam Dome Pressure—Low is one of the Functions assumed to be OPERABLE and capable of permitting initiation of the ECCS during the transients analyzed in References 1 and 3. In addition, the Reactor Steam Dome Pressure—Low Function is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The Reactor Steam Dome Pressure—Low signals are initiated from four pressure transmitters that sense the reactor dome pressure.

The Allowable Value is low enough to prevent overpressurizing the equipment in the low pressure ECCS, but high enough to ensure that the ECCS injection prevents the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46.

Four channels of Reactor Steam Dome Pressure—Low Function are only required to be OPERABLE when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

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BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

1.d. Core Spray Pump Discharge Flow- Low (Bypass)

The minimum flow instruments are provided to protect the associated CS pumps from overheating when the pump is operating and the associated injection valve is not fully open. The minimum flow line valve is opened when low flow is sensed, and the valve is automatically closed when the flow rate is adequate to protect the pump. The CS Pump Discharge Flow-Low Function is assumed to be OPERABLE and capable of closing the minimum flow valves to ensure that the CS flows assumed during the transients and accidents analyzed in References 1, 2, and 3 are met. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

One flow switch per CS subsystem is used to detect the associated subsystems' flow rates. The logic is arranged such that each flow switch causes its associated minimum flow valve to open. The logic will close the minimum flow valve once the closure setpoint is exceeded. The Pump Discharge Flow-Low Allowable Values are high enough to ensure that the pump flow rate is sufficient to protect the pump, yet low enough (based on engineering judgment) to ensure that the closure of the minimum flow valve is initiated to allow full flow into the core.

Each channel of Pump Discharge Flow-Low Function (two CS channels) is only required to be OPERABLE when the associated ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude the ECCS function. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

1.e, 2.f. Core Spray and Low Pressure Coolant Injection
Pump Start- Time Delay Relay

The reaction of the low pressure ECCS pumps to an initiation signal depends on the availability of power. If normal power (offsite power) is not available, the four RHR (LPCI) pumps start simultaneously after the standby power source (four diesel generators) is available while the CS pumps start simultaneously after a seven-second time delay. This

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BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

1.e, 2.f. Core Spray and Low Pressure Coolant Injection
Pump Start-Time Delay Relay (continued)

time delay allows the start of LPCI pumps to avoid overloading the diesel generators. When normal power is available, the CS and RHR pump starts are staggered by shutdown board (i.e., A pumps start at 0 seconds, B pumps start at 7 seconds, C pumps start at 14 seconds, and D pumps start at 21 seconds). The purpose of this time delay, when power is being provided from the normal power source (offsite), is to stagger the start of the CS and LPCI pumps, thus limiting the starting transients on the 4.16 kV shutdown buses. The CS and LPCI Pump Start-Time Delay Relays are assumed to be OPERABLE in the accident and transient analyses requiring ECCS initiation. That is, the analyses assume that the pumps will initiate when required and excess loading will not cause failure of the power sources.

There are four CS Pump and six LPCI Pump Start-Time Delay Relays when power is being provided from the normal power source, one in each of the pump start logic circuits (LPCI pumps C and D have two time delay relays). While each time delay relay is dedicated to a single pump start logic, a single failure of a CS or LPCI Pump Start-Time Delay Relay could result in the loss of normal power to a 4.16 kV shutdown board due to a voltage transient on the associated shutdown bus (e.g., as in the case where ECCS pumps on one shutdown bus start simultaneously due to an inoperable time delay relay). This would result in the affected board being powered by the associated diesel. Therefore, the worst case single failure would be failure of a single pump to start due to a relay failure leaving seven of the eight low pressure ECCS pumps OPERABLE; thus, the single failure criterion is met (i.e., loss of one instrument does not preclude ECCS initiation). Since the CS pumps are 50% capacity pumps, the LOCA analysis does not take credit for a CS loop if one of the pumps is inoperable. Therefore, a 4.16 kV shutdown board failure results in the loss of one RHR pump and one CS loop (two CS pumps) for the LOCA analysis. The Allowable Value for the CS and LPCI Pump

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BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1.e, 2.f. Core Spray and Low Pressure Coolant Injection
Pump Start-Time Delay Relay (continued)

Start-Time Delay Relays is chosen to be long enough so that most of the starting transient of the first set of pumps is complete before starting the second set of pumps on the same 4.16 kV shutdown bus and short enough so that ECCS operation is not degraded.

There are also four CS and six LPCI Pump Start-Time Delay Relays when power is being provided by the standby source, one in each of the pump start logic circuits (LPCI pumps C and D have two time delay relays). While each relay is dedicated to a single pump start logic, a single failure of a Pump Start-Time Delay Relay could result in the failure of the two low pressure ECCS pumps (CS and LPCI) powered from the same shutdown board, to perform their intended function (e.g., as in the case where both ECCS pumps on one shutdown board start simultaneously due to an inoperable time delay relay). This still leaves six of eight low pressure ECCS pumps OPERABLE; thus, the single failure criterion is met (i.e., loss of one instrument does not preclude ECCS initiation). As stated above, since the LOCA analysis does not take credit for a CS loop if one of the pumps is inoperable, the loss of a 4.16 kV shutdown board effectively results in the loss of one LPCI pump and one CS loop (two CS pumps). The Allowable Value for the CS and LPCI Pump Start-Time Delay Relays is chosen to be long enough so that most of the starting transient for the LPCI pump is complete before starting the CS pump on the same 4.16 kV shutdown board and short enough so that ECCS operation is not degraded.

Each CS and LPCI Pump Start-Time Delay Relay Function is required to be OPERABLE only when the associated CS and LPCI subsystems are required to be OPERABLE. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the CS and LPCI subsystems.

2.d. Reactor Steam Dome Pressure-Low (Recirculation
Discharge Valve Permissive)

Low reactor steam dome pressure signals are used as permissives for recirculation discharge valve closure. This ensures that the LPCI subsystems inject into the proper RPV

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BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

2.d. Reactor Steam Dome Pressure—Low (Recirculation
Discharge Valve Permissive) (continued)

location assumed in the safety analysis. The Reactor Steam Dome Pressure—Low is one of the Functions assumed to be OPERABLE and capable of closing the valve during the transients analyzed in References 1 and 3. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Steam Dome Pressure—Low Function is directly assumed in the analysis of the recirculation line break (Ref. 2).

The Reactor Steam Dome Pressure—Low signals are initiated from four pressure transmitters that sense the reactor dome pressure.

The Allowable Value is chosen to ensure that the valves close prior to commencement of LPCI injection flow into the core, as assumed in the safety analysis.

Four channels of the Reactor Steam Dome Pressure—Low Function are only required to be OPERABLE in MODES 1, 2, and 3 with the associated recirculation pump discharge valve open. With the valve(s) closed, the function of the instrumentation has been performed; thus, the Function is not required. In MODES 4 and 5, the loop injection location is not critical since LPCI injection through the recirculation loop in either direction will still ensure that LPCI flow reaches the core (i.e., there is no significant reactor steam dome back pressure).

2.e. Reactor Vessel Water Level—Level 0

The Level 0 Function is provided as a permissive to allow the RHR System to be manually aligned from the LPCI mode to the suppression pool cooling/spray or drywell spray modes. The permissive ensures that water in the vessel is approximately two thirds core height before the manual transfer is allowed. This ensures that LPCI is available to prevent or minimize fuel damage. This function may be overridden during accident conditions as allowed by plant procedures. Reactor Vessel Water Level—Level 0 Function is implicitly assumed in the analysis of the recirculation line break (Ref. 2) since the analysis assumes that no LPCI flow diversion occurs when reactor water level is below Level 0.

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BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

2.e. Reactor Vessel Water Level - Level 0 (continued)

Reactor Vessel Water Level - Level 0 signals are initiated from two level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. The Reactor Vessel Water Level - Level 0 Allowable Value is chosen to allow the low pressure core flooding systems to activate and provide adequate cooling before allowing a manual transfer.

Two channels of the Reactor Vessel Water Level - Level 0 Function are only required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the specified initiation time of the LPCI subsystems is not assumed, and other administrative controls are adequate to control the valves that this Function isolates (since the systems that the valves are opened for are not required to be OPERABLE in MODES 4 and 5 and are normally not used).

HPCI System

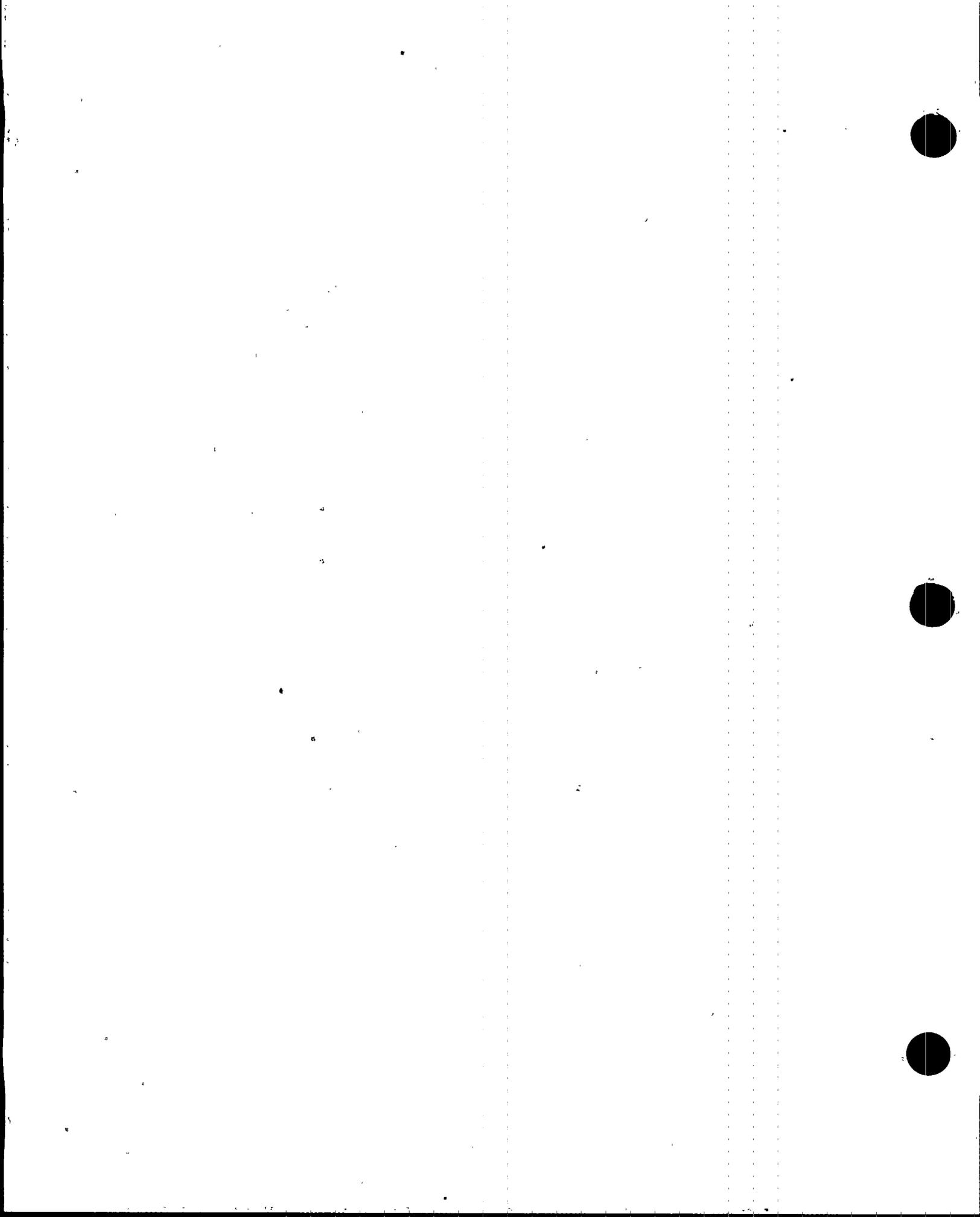
3.a. Reactor Vessel Water Level - Low Low, Level 2

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the HPCI System is initiated at Level 2 to maintain level above the top of the active fuel. The Reactor Vessel Water Level - Low Low, Level 2 is one of the Functions assumed to be OPERABLE and capable of initiating HPCI during the transients analyzed in References 1, 2, and 3. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low Low, Level 2 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low, Level 2 Allowable Value is high enough such that for complete loss of

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BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3.a. Reactor Vessel Water Level - Low Low, Level 2
(continued)

feedwater flow, the Reactor Core Isolation Cooling (RCIC) System flow with HPCI assumed to fail will be sufficient to avoid initiation of low pressure ECCS at Reactor Vessel Water Level - Low Low Low, Level 1.

Four channels of Reactor Vessel Water Level - Low Low, Level 2 Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.b. Drywell Pressure - High

High pressure in the drywell could indicate a break in the RCPB. The HPCI System is initiated upon receipt of the Drywell Pressure - High Function in order to minimize the possibility of fuel damage. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible to be indicative of a LOCA inside primary containment.

Four channels of the Drywell Pressure - High Function are required to be OPERABLE when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.c. Reactor Vessel Water Level - High, Level 8

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Level 8 signal is used to trip the HPCI turbine to prevent overflow into the main steam lines (MSLs). The Reactor Vessel Water Level - High, Level 8 Function is not assumed in the accident and transient analyses. It was retained since it is a

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3.c. Reactor Vessel Water Level - High, Level 8
(continued)

potentially significant contributor to risk, thus it meets Criterion 4 of the NRC Policy Statement (Ref. 5).

Reactor Vessel Water Level - High, Level 8 signals for HPCI are initiated from two level transmitters from the narrow range water level measurement instrumentation. The Reactor Vessel Water Level - High, Level 8 Allowable Value is chosen to prevent flow from the HPCI System from overflowing into the MSIs.

Two channels of Reactor Vessel Water Level - High, Level 8 Function are required to be OPERABLE only when HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.d. Condensate Header Level - Low

Low level in the CST indicates the unavailability of an adequate supply of makeup water from this normal source. Normally the suction valves between HPCI and the CST are open and, upon receiving a HPCI initiation signal, water for HPCI injection would be taken from the CST. However, if the water level in the HPCI pump supply header from the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the HPCI pump. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes. The Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool.

Condensate Header Level - Low signals are initiated from two level switches. The logic is arranged such that either level switch can cause the suppression pool suction valves to open and the CST suction valve to close. The Condensate Header Level - Low Function Allowable Value is high enough to ensure adequate pump suction head while water is being taken from the CST.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3.d. Condensate Header Level - Low (continued)

One channel of the Condensate Header Level - Low Function is required to be OPERABLE only when HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.e. Suppression Pool Water Level - High

Excessively high suppression pool water could result in the loads on the suppression pool exceeding design values should there be a blowdown of the reactor vessel pressure through the safety/relief valves. Therefore, signals indicating high suppression pool water level are used to transfer the suction source of HPCI from the CST to the suppression pool to eliminate the possibility of HPCI continuing to provide additional water from a source outside containment. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes.

This Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool.

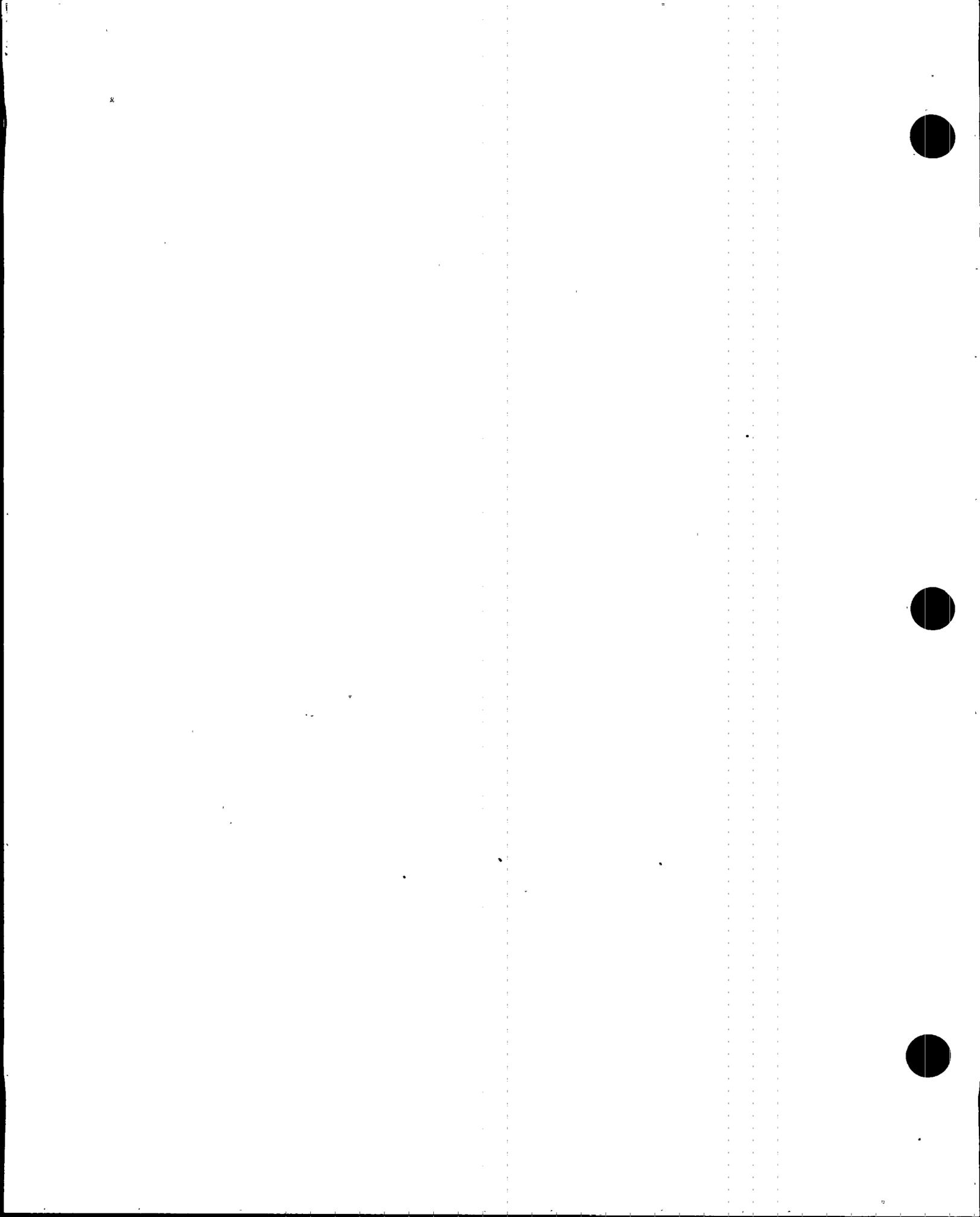
Suppression Pool Water Level - High signals are initiated from two level switches. The logic is arranged such that either switch can cause the suppression pool suction valves to open and the CST suction valve to close. The Allowable Value for the Suppression Pool Water Level - High Function is chosen to ensure that HPCI will be aligned for suction from the suppression pool before the water level reaches the point at which suppression pool design loads would be exceeded.

One channel of Suppression Pool Water Level - High Function is required to be OPERABLE only when HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.f. High Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)

The minimum flow instruments are provided to protect the HPCI pump from overheating when the pump is operating at reduced flow. The minimum flow line valve is opened when low flow is sensed, and the valve is automatically closed

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3.f. High Pressure Coolant Injection Pump Discharge
Flow-Low (Bypass) (continued)

when the flow rate is adequate to protect the pump. The High Pressure Coolant Injection Pump Discharge Flow-Low Function is assumed to be OPERABLE and capable of closing the minimum flow valve to ensure that the ECCS flow assumed during the transients and accidents analyzed in References 2 and 3 are met. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

One flow transmitter is used to detect the HPCI System's flow rate. The logic is arranged such that the transmitter causes the minimum flow valve to open. The logic will close the minimum flow valve once the closure setpoint is exceeded.

The High Pressure Coolant Injection Pump Discharge Flow-Low Allowable Value is high enough to ensure that pump flow rate is sufficient to protect the pump, yet low enough (based on engineering judgment) to ensure that the closure of the minimum flow valve is initiated to allow full flow into the core.

One channel is required to be OPERABLE when the HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

Automatic Depressurization System

4.a, 5.a. Reactor Vessel Water Level-Low Low Low, Level 1

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, ADS receives one of the signals necessary for initiation from this Function. The Reactor Vessel Water Level-Low Low Low, Level 1 is one of the Functions assumed to be OPERABLE and capable of initiating the ADS during the accident analyzed in Reference 2. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

4.a, 5.a. Reactor Vessel Water Level - Low Low Low, Level 1
(continued)

Reactor Vessel Water Level - Low Low Low, Level 1 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Low Low, Level 1 Function are required to be OPERABLE only when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

The Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value is chosen to allow time for the low pressure core flooding systems to initiate and provide adequate cooling.

4.b, 5.b. Drywell Pressure - High

High pressure in the drywell could indicate a break in the RCPB. Therefore, ADS receives one of the signals necessary for initiation from this Function in order to minimize the possibility of fuel damage. The Drywell Pressure - High is assumed to be OPERABLE and capable of initiating the ADS during the accidents analyzed in Reference 2. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Drywell Pressure - High signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

Four channels of Drywell Pressure - High Function are only required to be OPERABLE when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

4.c, 5.c. Automatic Depressurization System Initiation
Timer

The purpose of the Automatic Depressurization System Initiation Timer is to delay depressurization of the reactor vessel to allow the HPCI System time to maintain reactor vessel water level. Since the rapid depressurization caused by ADS operation is one of the most severe transients on the reactor vessel, its occurrence should be limited. By delaying initiation of the ADS Function, the operator is given the chance to monitor the success or failure of the HPCI System to maintain water level, and then to decide whether or not to allow ADS to initiate, to delay initiation further by recycling the timer, or to inhibit initiation permanently. The Automatic Depressurization System Initiation Timer Function is assumed to be OPERABLE for the accident analyses of Reference 2 that require ECCS initiation.

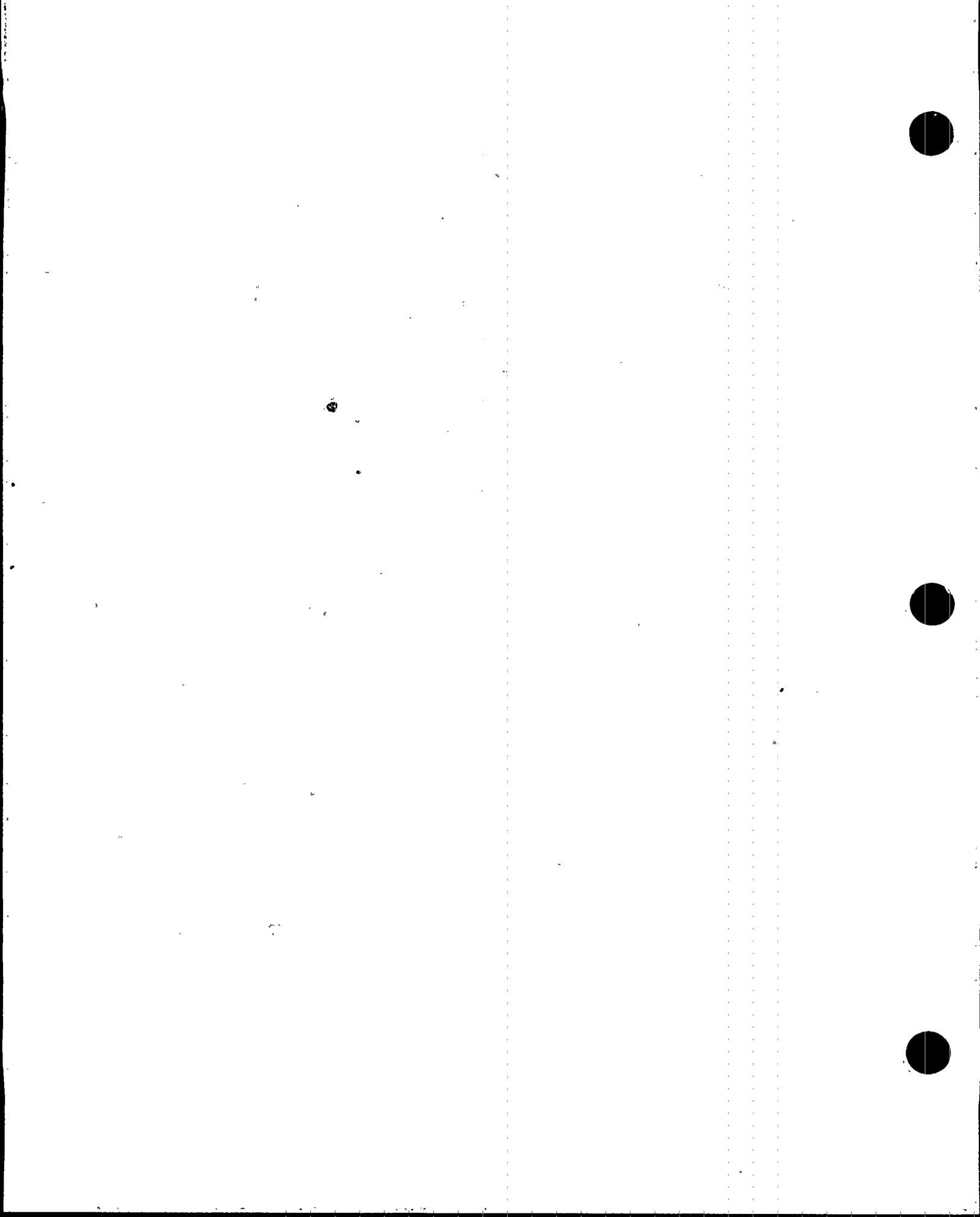
There are two Automatic Depressurization System Initiation Timer relays, one in each of the two ADS trip systems. The Allowable Value for the Automatic Depressurization System Initiation Timer is chosen so that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

Two channels of the Automatic Depressurization System Initiation Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. (One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.d, 5.d. Reactor Vessel Water Level - Low, Level 3
(Confirmatory)

The Reactor Vessel Water Level - Low, Level 3 (Confirmatory) Function is used by the ADS only as a confirmatory low water level signal. ADS receives one of the signals necessary for initiation from Reactor Vessel Water Level - Low Low Low, Level 1 signals. In order to prevent spurious initiation of the ADS due to spurious Level 1 signals, a Level 3 (Confirmatory) signal must also be received before ADS initiation commences.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

4.d, 5.d. Reactor Vessel Water Level - Low, Level 3
(Confirmatory) (continued)

Reactor Vessel Water Level - Low, Level 3 (Confirmatory) signals are initiated from two level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

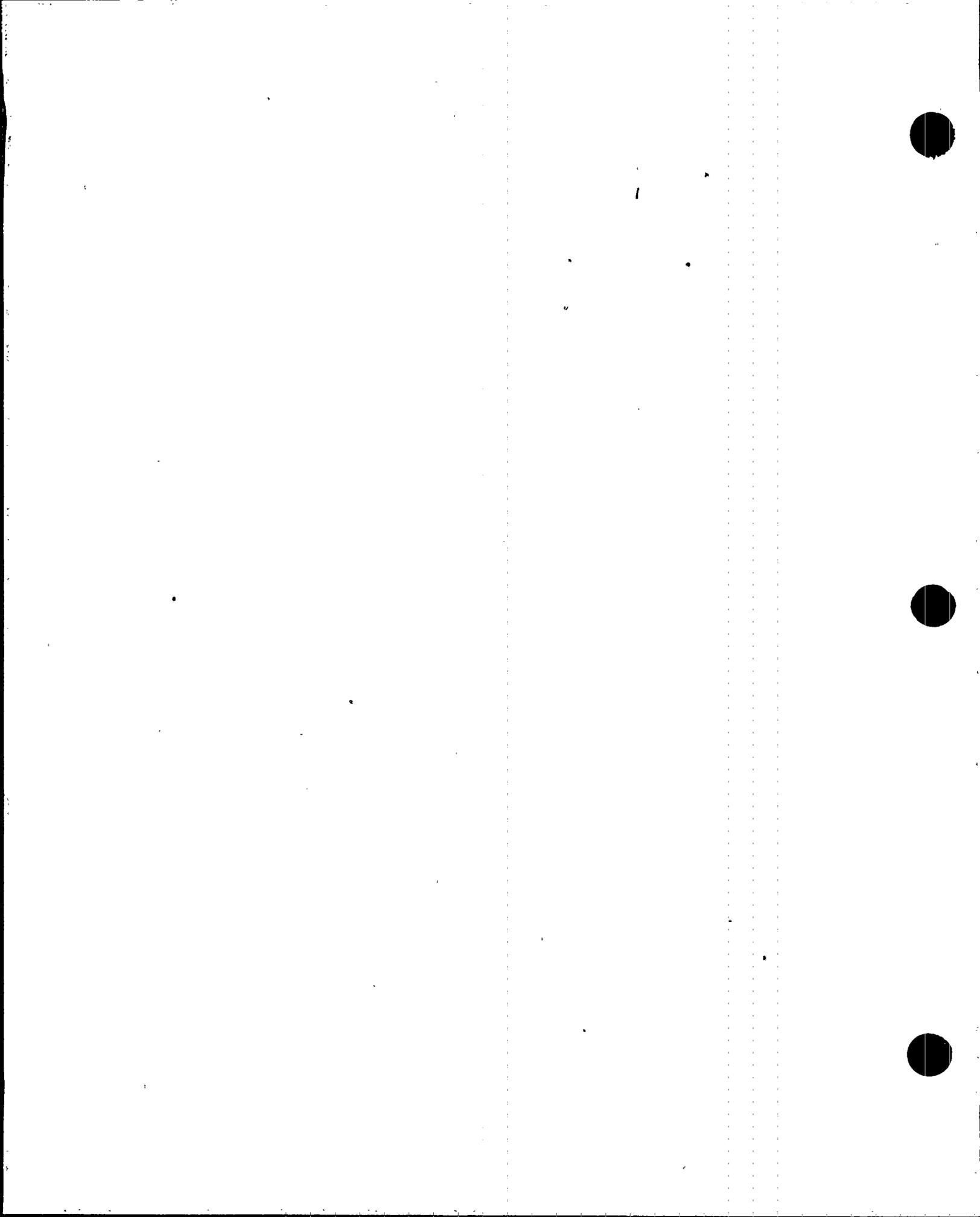
Two channels of Reactor Vessel Water Level - Low, Level 3 (Confirmatory) Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.e, 4.f, 5.e, 5.f. Core Spray and Low Pressure Coolant
Injection Pump Discharge Pressure - High

The Pump Discharge Pressure - High signals from the CS and LPCI pumps are used as permissives for ADS initiation, indicating that there is a source of low pressure cooling water available once the ADS has depressurized the vessel. Pump Discharge Pressure - High is one of the Functions assumed to be OPERABLE and capable of permitting ADS initiation during the events analyzed in Reference 2 with an assumed HPCI failure. For these events the ADS depressurizes the reactor vessel so that the low pressure ECCS can perform the core cooling functions. This core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Pump discharge pressure signals are initiated from twelve pressure transmitters, two on the discharge side of each RHR (LPCI) pump and one on the discharge side of each CS pump. There are two ADS low pressure ECCS pump permissives in each trip system. Each of these permissives receives inputs from all four RHR (LPCI) pumps (different signals for each permissive) and two CS pumps, two for each subsystem (different pumps for each permissive). In order to generate an ADS permissive in one trip system, it is necessary that

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

4.e, 4.f, 5.e, 5.f. Core Spray and Low Pressure Coolant
Injection Pump Discharge Pressure-High (continued)

only one LPCI pump or two CS pumps (CS pumps A or B and either C or D) indicate the high discharge pressure condition. The Pump Discharge Pressure-High Allowable Value is less than the pump discharge pressure when the pump is operating in a full flow mode and high enough to avoid any condition that results in a discharge pressure permissive when the CS and LPCI pumps are aligned for injection and the pumps are not running. The actual operating point of this function is not assumed in any transient or accident analysis. However, this function is indirectly assumed to operate (in Reference 2) to provide the ADS permissive to depressurize the RCS to allow the ECCS low pressure systems to operate.

Twelve channels of Core Spray and Low Pressure Coolant Injection Pump Discharge Pressure-High Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Four CS channels associated with CS pumps A through D and eight LPCI channels associated with LPCI pumps A through D are required for trip systems. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.g, 5.g. Automatic Depressurization System High Drywell
Pressure Bypass Timer

One of the signals required for ADS initiation is Drywell Pressure-High. However, if the event requiring ADS initiation occurs outside the drywell (e.g., main steam line break outside containment), a high drywell pressure signal may never be present. Therefore, the Automatic Depressurization System High Drywell Pressure Bypass Timer is used to bypass the Drywell Pressure-High Function after a certain time period has elapsed. Operation of the Automatic Depressurization System High Drywell Pressure Bypass Timer Function is not assumed in any accident analysis. The instrumentation was installed to meet requirements of NUREG-0737, Item II.K.3.18 (Ref. 6) and is retained in the TS because ADS is part of the primary success path for mitigation of a DBA.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

4.g, 5.g. Automatic Depressurization System High Drywell
Pressure Bypass Timer (continued)

There are two Automatic Depressurization System High Drywell Pressure Bypass Timer relays, one in each of the two ADS trip systems. The Allowable Value for the Automatic Depressurization System High Drywell Pressure Bypass Timer is chosen to ensure that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

Two channels of the Automatic Depressurization System High Drywell Pressure Bypass Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Refer to LCO 3.5.1 for ADS Applicability Bases.

ACTIONS

A Note has been provided to modify the ACTIONS related to ECCS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable ECCS instrumentation channels provide appropriate compensatory measures for separate inoperable Condition entry for each inoperable ECCS instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.5.1-1. The applicable Condition referenced in the table is Function dependent. Each time a channel is discovered inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

(continued)



BASES

ACTIONS
(continued)B.1, B.2, and B.3

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in redundant automatic initiation capability being lost for the feature(s). Required Action B.1 features would be those that are initiated by Functions 1.a, 1.b, 1.c, 2.a, 2.b, and 2.c (e.g., low pressure ECCS). The Required Action B.2 system would be HPCI. For Required Action B.1, redundant automatic initiation capability is lost if (a) two or more Function 1.a channels are inoperable and untripped such that both trip systems lose initiation capability, (b) two or more Function 2.a channels are inoperable and untripped such that both trip systems lose initiation capability, (c) two or more Function 1.b channels are inoperable and untripped such that both trip systems lose initiation capability, (d) two or more Function 2.b channels are inoperable and untripped such that both trip systems lose initiation capability, (e) two or more Function 1.c channels are inoperable and untripped such that both trip systems lose initiation capability, or (f) two or more Function 2.c channels are inoperable and untripped such that both trip systems lose initiation capability. For low pressure ECCS, since each inoperable channel would have Required Action B.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated system of low pressure ECCS, DGs, and EECW to be declared inoperable. However, since channels in both associated low pressure ECCS subsystems (e.g., both CS subsystems) are inoperable and untripped, and the Completion Times started concurrently for the channels in both subsystems, this results in the affected portions in the associated low pressure ECCS, DGs, and EECW being concurrently declared inoperable.

For Required Action B.2, redundant automatic HPCI initiation capability is lost if two or more Function 3.a or two or more Function 3.b channels are inoperable and untripped such that the trip system loses initiation capability. In this situation (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action B.3 is not appropriate and the HPCI System must be declared inoperable within 1 hour. As noted (Note 1 to Required Action B.1), Required Action B.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific

(continued)



BASES

ACTIONS

B.1, B.2, and B.3 (continued)

initiation time of the low pressure ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of initiation capability for 24 hours (as allowed by Required Action B.3) is allowed during MODES 4 and 5. There is no similar Note provided for Required Action B.2 since HPCI instrumentation is not required in MODES 4 and 5; thus, a Note is not necessary.

Notes are also provided (Note 2 to Required Action B.1 and the Note to Required Action B.2) to delineate which Required Action is applicable for each Function that requires entry into Condition B if an associated channel is inoperable. This ensures that the proper loss of initiation capability check is performed. Required Action B.1 (the Required Action for certain inoperable channels in the low pressure ECCS subsystems) is not applicable to Function 2.e, since this Function provides backup to administrative controls ensuring that operators do not divert LPCI flow from injecting into the core when needed. Thus, a total loss of Function 2.e capability for 24 hours is allowed, since the LPCI subsystems remain capable of performing their intended function.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action B.1, the Completion Time only begins upon discovery that a redundant feature in the same system (e.g., both CS subsystems) cannot be automatically initiated due to inoperable, untripped channels within the same Function as described in the paragraph above. For Required Action B.2, the Completion Time only begins upon discovery that the HPCI System cannot be automatically initiated due to two inoperable, untripped channels for the associated Function in the same trip system. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 4) to permit restoration of any

(continued)



BASES

ACTIONS

B.1, B.2, and B.3 (continued)

inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.3. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition H must be entered and its Required Action taken.

C.1 and C.2

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the same Function result in redundant automatic initiation capability being lost for the feature(s). Required Action C.1 features would be those that are initiated by Functions 1.e, 2.d, and 2.f (i.e., low pressure ECCS). Redundant automatic initiation capability is lost if either (a) two or more Function 1.e channels are inoperable affecting CS pumps in different subsystems, (b) two or more Function 2.d channels are inoperable in the same trip system such that the trip system loses initiation capability, or (c) two or more or more Function 2.f channels are inoperable affecting two LPCI pumps. In this situation (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action C.2 is not appropriate and the feature(s) associated with the inoperable channels must be declared inoperable within 1 hour. Since each inoperable channel would have Required Action C.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated system to be declared inoperable. However, since channels for both low pressure ECCS subsystems are inoperable (e.g., both CS subsystems), and the Completion Times started concurrently for the channels in both subsystems, this results in the affected portions in both subsystems being concurrently declared inoperable. For Functions 1.e, 2.d, and 2.f, the affected portions are the associated low pressure ECCS pumps. As noted (Note 1), Required Action C.1 is only applicable in MODES 1, 2, and 3.

(continued)



BASES

ACTIONS

C.1 and C.2 (continued)

In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of automatic initiation capability for 24 hours (as allowed by Required Action C.2) is allowed during MODES 4 and 5.

Note 2 states that Required Action C.1 is only applicable for Functions 1.e, 2.d, and 2.f. Required Action C.1 is also not applicable to Function 3.c (which also requires entry into this Condition if a channel in this Function is inoperable), since the loss of one channel results in a loss of the Function (two-out-of-two logic). This loss was considered during the development of Reference 4 and considered acceptable for the 24 hours allowed by Required Action C.2.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action C.1, the Completion Time only begins upon discovery that the same feature in both subsystems (e.g., both CS subsystems) cannot be automatically initiated due to inoperable channels within the same Function as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would either cause the initiation or it would not necessarily result in a safe state for the channel in all events.

(continued)



BASES

ACTIONS
(continued)

D.1

Required Action D.1 is intended to ensure that appropriate actions are taken if an inoperable, untripped channel within the same Function results in a complete loss of automatic component initiation capability for the HPCI System. Since Table 3.3.5.1-1 only requires one channel to be OPERABLE, automatic component initiation capability is lost if the one required Function 3.d channel or the one required Function 3.e channel is inoperable and untripped. In this situation (loss of automatic suction swap), the HPCI system must be declared inoperable within 1 hour. As noted, Required Action D.1 is only applicable if the HPCI pump suction is not aligned to the suppression pool, since, if aligned, the Function is already performed.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action D.1, the Completion Time only begins upon discovery that the HPCI System cannot be automatically aligned to the suppression pool due to an inoperable, untripped channels in the same Function. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

E.1 and E.2

Required Action E.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the Core Spray Pump Discharge Flow-Low Bypass Function results in redundant automatic initiation capability being lost for the feature(s). For Required Action E.1, the features would be those that are initiated by Function 1.d (i.e., CS). Redundant automatic initiation capability is lost if two Function 1.d channels are inoperable.

In this situation (loss of minimum flow capability), the 7 day allowance of Required Action E.2 is not appropriate and the subsystem associated with each inoperable channel must be declared inoperable within 1 hour. As noted (Note 1 to Required Action E.1), Required Action E.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the

(continued)



BASES

ACTIONS

E.1 and E.2 (continued)

specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of initiation capability for 7 days (as allowed by Required Action E.2) is allowed during MODES 4 and 5. A Note is also provided (Note 2 to Required Action E.1) to delineate that Required Action E.1 is only applicable to low pressure ECCS Functions. Required Action E.1 is not applicable to HPCI Function 3.f since the loss of one channel results in a loss of the Function (one-out-of-one logic). This loss was considered during the development of Reference 4 and considered acceptable for the 7 days allowed by Required Action E.2.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock."

For Required Action E.1, the Completion Time only begins upon discovery that a redundant feature in the same system (e.g., both CS subsystems) cannot be automatically initiated due to inoperable channels within the same Function as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

If the instrumentation that controls the CS pump minimum flow valve is inoperable, such that the valve will not automatically open, extended CS pump operation with no injection path available could lead to pump overheating and failure. If there were a failure of the instrumentation, such that the valve would not automatically close, a portion of the pump flow could be diverted from the reactor vessel injection path, causing insufficient core cooling. These consequences can be averted by the operator's manual control of the valve, which would be adequate to maintain ECCS pump protection and required flow. Furthermore, other ECCS pumps would be sufficient to complete the assumed safety function if no additional single failure were to occur. The 7 day Completion Time of Required Action E.2 to restore the inoperable channel to OPERABLE status is reasonable based on the remaining capability of the associated ECCS subsystems, the redundancy available in the ECCS design, and the low

(continued)



BASES

ACTIONS

E.1 and E.2 (continued)

probability of a DBA occurring during the allowed out of service time. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

F.1 and F.2

Required Action F.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within similar ADS trip system A and B Functions result in redundant automatic initiation capability being lost for the ADS. Redundant automatic initiation capability is lost if either (a) one or more Function 4.a channels and one or more Function 5.a channels are inoperable and untripped, (b) one or more Function 4.b channels and one or more Function 5.b channels are inoperable and untripped, or (c) one or more Function 4.d channels and one or more Function 5.d channels are inoperable and untripped.

In this situation (loss of automatic initiation capability), the 96 hour or 8 day allowance, as applicable, of Required Action F.2 is not appropriate and all ADS valves must be declared inoperable within 1 hour after discovery of loss of ADS initiation capability.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action F.1, the Completion Time only begins upon discovery that the ADS cannot be automatically initiated due to inoperable, untripped channels within similar ADS trip system Functions as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an

(continued)



BASES

ACTIONS

F.1 and F.2 (continued)

allowable out of service time of 8 days has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status if both HPCI and RCIC are OPERABLE. If either HPCI or RCIC is inoperable, the time is shortened to 96 hours. If the status of HPCI or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCI or RCIC inoperability. However, the total time for an inoperable, untripped channel cannot exceed 8 days. If the status of HPCI or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable, untripped channel. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action F.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition H must be entered and its Required Action taken.

G.1 and G.2

Required Action G.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within similar ADS trip system Functions result in automatic initiation capability being lost for the ADS. Automatic initiation capability is lost if either (a) one Function 4.c channel and one Function 5.c channel are inoperable, (b) a combination of Function 4.e, 4.f, 5.e, and 5.f channels are inoperable such that channels associated with five or more low pressure ECCS pumps are inoperable, or (c) one or more Function 4.g channels and one or more Function 5.g channels are inoperable.

In this situation (loss of automatic initiation capability), the 96 hour or 8 day allowance, as applicable, of Required Action G.2 is not appropriate, and all ADS valves must be declared inoperable within 1 hour after discovery of loss of ADS initiation capability.

(continued)



BASES

ACTIONS

G.1 and G.2 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action G.1, the Completion Time only begins upon discovery that the ADS cannot be automatically initiated due to inoperable channels within similar ADS trip system Functions as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 8 days has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status if both HPCI and RCIC are OPERABLE (Required Action G.2). If either HPCI or RCIC is inoperable, the time shortens to 96 hours. If the status of HPCI or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCI or RCIC inoperability. However, the total time for an inoperable channel cannot exceed 8 days. If the status of HPCI or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable channel. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

H.1

With any Required Action and associated Completion Time not met, the associated feature(s) may be incapable of performing the intended function, and the supported feature(s) associated with inoperable untripped channels must be declared inoperable immediately.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

As noted in the beginning of the SRs, the SRs for each ECCS instrumentation Function are found in the SRs column of Table 3.3.5.1-1.

The Surveillances are modified by a second Note (Note 2) to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours as follows: (a) for Functions 3.c and 3.f; and (b) for Functions other than 3.c and 3.f provided the associated Function or redundant Function maintains ECCS initiation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 4) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the ECCS will initiate when necessary.

SR 3.3.5.1.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.5.1.1 (continued)

supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.5.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analyses of Reference 4.

SR 3.3.5.1.3, SR 3.3.5.1.4, and SR 3.3.5.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequencies of SR 3.3.5.1.3, SR 3.3.5.1.4, and SR 3.3.5.1.5 are based upon the magnitude of equipment drift in the setpoint analysis.

SR 3.3.5.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.7.2, LCO 3.8.1, and LCO 3.8.2 overlaps this Surveillance to complete testing of the assumed safety function.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.5.1.6 (continued)

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. FSAR, Section 8.5.
 2. FSAR, Section 6.5.
 3. FSAR, Chapter 14.
 4. NEDC-30936-P-A, "BWR Owners' Group Technical Specification Improvement Analyses for ECCS Actuation Instrumentation, Part 2," December 1988.
 5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
 6. NUREG-0737, "Clarification of TMI Action Plan Requirements," October 31, 1980.
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B 3.3 INSTRUMENTATION

B 3.3.5.2 Reactor Core Isolation Cooling (RCIC) System Instrumentation

BASES

BACKGROUND

The purpose of the RCIC System instrumentation is to initiate actions to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink (the main condenser) and normal coolant makeup flow from the Reactor Feedwater System is unavailable, such that initiation of the low pressure Emergency Core Cooling Systems (ECCS) pumps does not occur. A more complete discussion of RCIC System operation is provided in the Bases of LCO 3.5.3, "RCIC System."

The RCIC System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of reactor vessel Low Low water level. The variable is monitored by four transmitters that are connected to four trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic arrangement. Once initiated, the RCIC logic seals in and can be reset by the operator only when the reactor vessel water level signals have cleared.

The RCIC test line isolation valve is closed on a RCIC initiation signal to allow full system flow.

There are two sources of water for RCIC operation. Reactor grade water in the CST is the normal source and the suppression pool is the alternate source. Although the RCIC System does not monitor the water levels in the High Pressure Coolant Injection (HPCI) supply header from the condensate storage tank (CST) and the suppression pool, administrative controls are in place that direct the transfer from the CST to the suppression pool when the HPCI System automatically transfers on low HPCI pump supply header level or high suppression pool level.

The RCIC System provides makeup water to the reactor until the reactor vessel water level reaches the high water level (Level 8) trip (two-out-of-two logic), at which time the RCIC steam supply closes and the minimum flow valve closes, if open. The RCIC System restarts if vessel level again drops to the low level initiation point (Level 2).

(continued)



BASES (continued)

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The function of the RCIC System to provide makeup coolant to the reactor is used to respond to transient events. The RCIC System is not an Engineered Safety Feature System and no credit is taken in the safety analyses for RCIC System operation. Based on its contribution to the reduction of overall plant risk, however, the system, and therefore its instrumentation meets Criterion 4 of the NRC Policy Statement (Ref. 2). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the RCIC System instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.5.2-1. Each Function must have a required number of OPERABLE channels with their setpoints within the specified Allowable Values, where appropriate. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint).

Allowable Values are specified for each RCIC System instrumentation Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Each Allowable Value specified accounts for instrument uncertainties appropriate to the Function. These uncertainties are described in the setpoint methodology.

The individual Functions are required to be OPERABLE in MODE 1, and in MODES 2 and 3 with reactor steam dome pressure > 150 psig since this is when RCIC is required to be OPERABLE. (Refer to LCO 3.5.3 for Applicability Bases for the RCIC System.)

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

1. Reactor Vessel Water Level - Low Low, Level 2

Low reactor pressure vessel (RPV) water level indicates that normal feedwater flow is insufficient to maintain reactor vessel water level and that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the RCIC System is initiated at Level 2 to assist in maintaining water level above the top of the active fuel.

Reactor Vessel Water Level - Low Low, Level 2 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low, Level 2 Allowable Value is set high enough such that for complete loss of feedwater flow, the RCIC System flow with high pressure coolant injection assumed to fail will be sufficient to avoid initiation of low pressure ECCS at Level 1.

Four channels of Reactor Vessel Water Level - Low Low, Level 2 Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC initiation. Refer to LCO 3.5.3 for RCIC Applicability Bases.

2. Reactor Vessel Water Level - High, Level 8

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Level 8 signal is used to close the RCIC steam supply valve to prevent overflow into the main steam lines (MSLs).

Reactor Vessel Water Level - High, Level 8 signals for RCIC are initiated from two level transmitters from the narrow range water level measurement instrumentation, which sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

2. Reactor Vessel Water Level—High, Level 8 (continued)

The Reactor Vessel Water Level—High, Level 8 Allowable Value is high enough to preclude closing the RCIC steam supply valve, yet low enough to trip the RCIC System prior to water overflowing into the MSLs.

Two channels of Reactor Vessel Water Level—High, Level 8 Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE. Refer to LCO 3.5.3 for RCIC Applicability Bases.

ACTIONS

A Note has been provided to modify the ACTIONS related to RCIC System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RCIC System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RCIC System instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.5.2-1. The applicable Condition referenced in the Table is Function dependent. Each time a channel is discovered to be inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

B.1 and B.2

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss

(continued)



BASES

ACTIONS

B.1 and B.2 (continued)

of automatic initiation capability for the RCIC System. In this situation (loss of automatic initiation capability), the 24 hour allowance of Required Action B.2 is not appropriate, and the RCIC System must be declared inoperable within 1 hour after discovery of loss of RCIC initiation capability.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action B.1, the Completion Time only begins upon discovery that the RCIC System cannot be automatically initiated due to two or more inoperable, untripped Reactor Vessel Water Level-Low Low, Level 2 channels such that the trip system loses initiation capability. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the redundancy of sensors available to provide initiation signals and the fact that the RCIC System is not assumed in any accident or transient analysis, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 1) to permit restoration of any inoperable channel to OPERABLE status. For conservatism, in some transient analyses, RCIC flow rates were used rather than HPCI flow rates. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition D must be entered and its Required Action taken.

C.1

A risk based analysis was performed and determined that an allowable out of service time of 24 hours (Ref. 1) is

(continued)



BASES

ACTIONS

C.1 (continued)

acceptable to permit restoration of any inoperable channel to OPERABLE status (Required Action C.1). A Required Action (similar to Required Action B.1) limiting the allowable out of service time, if a loss of automatic RCIC initiation capability exists, is not required. This Condition applies to the Reactor Vessel Water Level-High, Level 8 Function whose logic is arranged such that any inoperable channel will result in a loss of automatic RCIC initiation capability. As stated above, this loss of automatic RCIC initiation capability was analyzed and determined to be acceptable. The Required Action does not allow placing a channel in trip since this action would not necessarily result in a safe state for the channel in all events.

D.1

With any Required Action and associated Completion Time not met, the RCIC System may be incapable of performing the intended function, and the RCIC System must be declared inoperable immediately.

SURVEILLANCE
REQUIREMENTS

As noted in the beginning of the SRs, the SRs for each RCIC System instrumentation Function are found in the SRs column of Table 3.3.5.2-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 6 hours for Function 2; and (b) for up to 6 hours for Function 1, provided the associated Function maintains trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 1) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the RCIC will initiate when necessary.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.5.2.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a parameter on other similar channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.5.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 1.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.5.2.3

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of SR 3.3.5.2.3 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.5.2.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.3 overlaps this Surveillance to provide complete testing of the safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. GENE-770-06-2, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.
 2. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.3 INSTRUMENTATION

B 3.3.6.1 Primary Containment Isolation Instrumentation

BASES

BACKGROUND

The primary containment isolation instrumentation automatically initiates closure of appropriate primary containment isolation valves (PCIVs). The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs). Primary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary containment and reactor coolant pressure boundary (RCPB) isolation. Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a primary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logics are (a) reactor vessel water level, (b) area ambient temperatures, (c) main steam line (MSL) flow measurement, (d) Standby Liquid Control (SLC) System initiation, (e) main steam line pressure, (f) high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) steam line flow, (g) drywell pressure, (h) HPCI and RCIC steam line pressure, (i) HPCI and RCIC turbine exhaust diaphragm pressure, and (j) reactor steam dome pressure. Redundant sensor input signals from each parameter are provided for initiation of isolation. The only exception is SLC System initiation.

Primary containment isolation instrumentation has inputs to the trip logic of the isolation functions listed below.

(continued)



BASES

BACKGROUND
(continued)

1. Main Steam Line Isolation

Most MSL Isolation Functions receive inputs from four channels. The outputs from these channels initiate isolation of the Group 1 isolation valves (main steam isolation valves (MSIVs) and MSL drain valves). The outputs from these channels are combined in a one-out-of-two taken twice logic to initiate isolation of the MSIVs. The outputs from the same channels are arranged into two two-out-of-two logic trip systems to isolate all MSL drain valves. The outputs from the Reactor Vessel Water Level - Low Low Low, Level 1 Function channels are arranged into two two-out-of-two logic trip systems to isolate the recirculation loop sample line valves.

The exceptions to this arrangement are the Main Steam Line Flow - High Function and Area Temperature Functions. The Main Steam Line Flow - High Function uses 16 flow channels, four for each steam line. One channel from each steam line inputs to each of the four trip strings. Two trip strings make up each trip system and both trip systems must trip to cause an MSL isolation. Each trip string has four inputs (one per MSL), any one of which will trip the trip string. The trip strings are arranged in a one-out-of-two taken twice logic.

The Main Steam Line Space Temperature - High Function receives input from 16 channels. The logic is arranged similar to the Main Steam Line Flow - High Function.

MSL Isolation Functions isolate the Group 1 valves.

2. Primary Containment Isolation

Most Primary Containment Isolation Functions receive inputs from four channels. The outputs from these channels are arranged into a one-out-of-two taken twice logic trip system. The isolation signal from this logic system isolates both containment isolation valves on a penetration.

Primary Containment Isolation Drywell Pressure - High and Reactor Vessel Water Level - Low, Level 3 Functions isolate the Group 2, 6 and 8 valves. The Reactor Vessel Water Level - Low, Level 3 Function also isolates Group 3 valves.

(continued)



BASES

BACKGROUND
(continued)

3, 4. High Pressure Coolant Injection System Isolation and
Reactor Core Isolation Cooling System Isolation

Most Functions that isolate HPCI and RCIC receive input from two channels, with each channel in one trip system using a one-out-of-two taken twice logic. Each of the two trip systems in each isolation group is connected to each of the two valves on each associated penetration.

The exceptions are the HPCI and RCIC Steam Line Flow-High Functions. There are two channels for this Function which provide an isolation signal to both trip systems using one-out-of-two logic. Each of the two trip systems isolate both valves in an associated penetration.

HPCI and RCIC Functions isolate the Group 4 and 5 valves.

5. Reactor Water Cleanup System Isolation

The Reactor Vessel Water Level-Low, Level 3 Isolation Function receives input from four reactor vessel water level channels. The outputs from the reactor vessel water level channels are connected into one-out-of-two taken twice trip systems. The SLC System Initiation Function provides an isolation signal to close both RWCU isolation valves. The Area Temperature-High Function receives input from twenty-four temperature monitors. There are four temperature sensors in each of the six areas where the RWCU piping and equipment are located. The four sensors in each area provide isolation signals to close both RWCU isolation valves using one-out-of-two logic.

RWCU Functions isolate the Group 3 valves. The Reactor Vessel Water Level-Low, Level 3 Function also isolates Group 2, 3, 6 and 8 valves.

6. Shutdown Cooling System Isolation

The Reactor Steam Dome Pressure-High Function receives input from two channels which provide one-out-of-two isolation logic to each isolation valve.

The Shutdown Cooling System Isolation Functions isolate the Group 2 RHR Shutdown Cooling (SDC) valves.

(continued)



BASES (continued)

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The isolation signals generated by the primary containment isolation instrumentation are implicitly assumed in the safety analyses of References 2 and 8 to initiate closure of valves to limit offsite doses. Refer to LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Applicable Safety Analyses Bases for more detail of the safety analyses.

Primary containment isolation instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 7). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the primary containment instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.6.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). Each channel must also respond within its assumed response time, where appropriate.

Allowable Values are specified for each Primary Containment Isolation Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
LCO, and
APPLICABILITY
(continued)

uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

Certain Emergency Core Cooling Systems (ECCS) and RCIC valves (e.g., minimum flow) also serve the dual function of automatic PCIVs. The signals that isolate these valves are also associated with the automatic initiation of the ECCS and RCIC. The instrumentation requirements and ACTIONS associated with these signals are addressed in LCO 3.3.5.1, "Emergency Core Cooling Systems (ECCS) Instrumentation," and LCO 3.3.5.2, "Reactor Core Isolation Cooling (RCIC) System Instrumentation," and are not included in this LCO. In general, the individual Functions are required to be OPERABLE in MODES 1, 2, and 3 consistent with the Applicability for LCO 3.6.1.1, "Primary Containment." Functions that have different Applicabilities are discussed below in the individual Functions discussion.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Main Steam Line Isolation

1.a. Reactor Vessel Water Level - Low Low Low, Level 1

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level - Low Low Low, Level 1 Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level - Low Low Low, Level 1 Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSLs on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg)

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
LCO, and
APPLICABILITY

1.a. Reactor Vessel Water Level — Low Low Low, Level 1
(continued)

and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low Low, Level 1 Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 100 limits.

This Function isolates the Group 1 valves.

1.b. Main Steam Line Pressure—Low

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure—Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

(continued)



BASES

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SAFETY ANALYSES,
LCO, and
APPLICABILITY

1.b. Main Steam Line Pressure—Low (continued)

The Main Steam Line Pressure—Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Flow—High

Main Steam Line Flow—High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow—High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 2).. The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits.

The MSL flow signals are initiated from 16 transmitters that are connected to the four MSLs. The transmitters are arranged such that, even though physically separated from each other, all four connected to one MSL would be able to detect the high flow. Four channels of Main Steam Line Flow—High Function for each MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break.

This Function isolates the Group 1 valves.

1.d. Main Steam Line Space Temperature—High

The Main Steam Line Space Temperature Function is provided to detect a leak in the RCPB and provides diversity to the high flow instrumentation. The isolation occurs when a very small leak has occurred. If the small leak is allowed to continue without isolation, offsite dose limits may be

(continued)



BASES

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LCO, and
APPLICABILITY

1.d. Main Steam Line Space Temperature—High (continued)

reached. However, credit for these instruments is not taken in any transient or accident analysis in the FSAR, since bounding analyses are performed for large breaks, such as MSLBs.

Main Steam Line Space temperature signals are initiated from bimetallic temperature switches located in the area being monitored. Sixteen channels of Main Steam Line Space Temperature—High Function are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The main steam line space temperature detection system Allowable Value is chosen to detect a leak equivalent to between 1% and 10% rated steam flow.

These Functions isolate the Group 1 valves.

Primary Containment Isolation

2.a. Reactor Vessel Water Level—Low, Level 3

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Reactor Vessel Water Level—Low, Level 3 Function associated with isolation is implicitly assumed in the FSAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level—Low, Level 3 signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low, Level 3 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

2.a. Reactor Vessel Water Level—Low, Level 3 (continued)

The Reactor Vessel Water Level—Low, Level 3 Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group 2, 3, 6, and 8 valves.

2.b. Drywell Pressure—High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Drywell Pressure—High Function, associated with isolation of the primary containment, is implicitly assumed in the FSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the ECCS Drywell Pressure—High Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA inside primary containment.

This Function isolates the Group 2, 6 and 8 valves.

High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems Isolation

3.a., 4.a. HPCI and RCIC Steam Line Flow—High

Steam Line Flow—High Functions are provided to detect a break of the RCIC or HPCI steam lines and initiate closure of the steam line isolation valves of the appropriate system. If the steam is allowed to continue flowing out of the break, the reactor will depressurize and the core can uncover. Therefore, the isolations are initiated on high flow to prevent or minimize core damage. The isolation

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3.a., 4.a. HPCI and RCIC Steam Line Flow—High
(continued)

action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. Specific credit for these Functions is not assumed in any FSAR accident analyses since the bounding analysis is performed for large breaks such as recirculation and MSL breaks. However, these instruments prevent the RCIC or HPCI steam line breaks from becoming bounding.

The HPCI and RCIC Steam Line Flow—High signals are initiated from transmitters (two for HPCI and two for RCIC) that are connected to the system steam lines. Two channels of both HPCI and RCIC Steam Line Flow—High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to be low enough to ensure that the trip occurs to prevent fuel damage and maintains the MSLB event as the bounding event.

These Functions isolate the Group 4 and 5 valves, as appropriate.

3.b., 4.b. HPCI and RCIC Steam Supply Line Pressure—Low

Low MSL pressure indicates that the pressure of the steam in the HPCI or RCIC turbine may be too low to continue operation of the associated system's turbine. These isolations are for equipment protection and are not assumed in any transient or accident analysis in the FSAR. However, they also provide a diverse signal to indicate a possible system break and provide the only signal which will isolate the steam supply lines for certain pipe breaks. These instruments are included in Technical Specifications (TS) because of the potential for risk due to possible failure of the instruments preventing HPCI and RCIC initiations. Therefore, they meet Criterion 4 of the NRC Policy Statement (Ref. 7).

The HPCI and RCIC Steam Supply Line Pressure—Low signals are initiated from switches (four for HPCI and four for RCIC) that are connected to the system steam line. Four

(continued)



BASES

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SAFETY ANALYSES,
LCO, and
APPLICABILITY

3.b., 4.b. HPCI and RCIC Steam Supply Line Pressure- Low
(continued)

channels of both HPCI and RCIC Steam Supply Line Pressure-Low Functions are available. Each Function is considered to have only one trip system since the output from the logic trips a common relay that initiates the isolations. Only three channels of each Function are required to be OPERABLE.

The Allowable Values are selected to be high enough to prevent damage to the system's turbine.

These Functions isolate the Group 4 and 5 valves, as appropriate.

3.c., 4.c. HPCI and RCIC Turbine Exhaust Diaphragm Pressure- High

High turbine exhaust diaphragm pressure indicates that the pressure may be too high to continue operation of the associated system's turbine. That is, one of two exhaust diaphragms has ruptured and pressure is reaching turbine casing pressure limits. These isolations are for equipment protection and are not assumed in any transient or accident analysis in the FSAR. These instruments are included in the TS because of the potential for risk due to possible failure of the instruments preventing HPCI and RCIC initiations. Therefore, they meet Criterion 4 of the NRC Policy Statement (Ref. 7).

The HPCI and RCIC Turbine Exhaust Diaphragm Pressure-High signals are initiated from switches (four for HPCI and four for RCIC) that are connected to the area between the rupture diaphragms on each system's turbine exhaust line. Four channels of both HPCI and RCIC Turbine Exhaust Diaphragm Pressure-High Functions are available. Each Function is considered to have only one trip system since the output from the logic trips a common relay that initiates the isolations. Only three channels of each Function are required to be OPERABLE.

The Allowable Values are low enough to prevent damage to the systems' turbines.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3.c., 4.c. HPCI and RCIC Turbine Exhaust Diaphragm
Pressure—High (continued)

These Functions isolate the Group 4 and 5 valves, as appropriate.

3.d., 3.e., 4.d., 4.e. Area Temperature—High

Area temperatures are provided to detect a leak from the associated system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. These Functions are not assumed in any FSAR transient or accident analysis, since bounding analyses are performed for large breaks such as recirculation or MSL breaks.

Area Temperature—High signals are initiated from bimetallic temperature switches that are appropriately located to protect the system that is being monitored. Four instruments monitor each area. Four channels for each HPCI and RCIC Area and Differential Temperature—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are set low enough to detect a leak equivalent to 25 gpm.

These Functions isolate the Group 4 and 5 valves, as appropriate.

Reactor Water Cleanup System Isolation

5.a., 5.b., 5.c., 5.d., 5.e., 5.f. Area Temperature—High

RWCU area temperatures are provided to detect a leak from the RWCU System. The isolation occurs even when very small leaks have occurred. If the small leak continues without isolation, offsite dose limits may be reached. Credit for these instruments is not taken in any transient or accident analysis in the FSAR, since bounding analyses are performed for large breaks such as recirculation or MSL breaks.

Area temperature signals are initiated from temperature elements that are located in the room that is being

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

5.a., 5.b., 5.c., 5.d, 5.e., 5.f. Area Temperature—High
(continued)

monitored. Four sensors in each area are required to be OPERABLE to provide isolation signals to close both RWCU isolation valves using one-out-of-two logic to ensure that no single instrument failure can preclude the isolation function.

The Area Temperature—High Allowable Values are set based on the maximum abnormal operating temperature for each area.

These Functions isolate the Group 3 valves.

5.g. SLC System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 4). An isolation signal for both RWCU isolation valves is initiated when the SLC pump start handswitch is not in the stop position.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

The SLC System Initiation Function is required to be OPERABLE only in MODES 1 and 2, since these are the only MODES where the reactor can be critical, and these MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

As noted (footnote (a) to Table 3.3.6.1-1), the SLC initiation signal provides input to the isolation logic for both RWCU isolation valves.

5.h. Reactor Vessel Water Level—Low, Level 3

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 3 supports actions to ensure that the fuel peak cladding temperature remains below the limits of

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

5.h. Reactor Vessel Water Level—Low, Level 3
(continued)

10 CFR 50.46. The Reactor Vessel Water Level—Low, Level 3 Function associated with RWCU isolation is not directly assumed in the FSAR safety analyses because the RWCU System line break is bounded by breaks of larger systems (recirculation and MSL breaks are more limiting).

Reactor Vessel Water Level—Low, Level 3 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low, Level 3 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

This Function isolates the Group 2, 3, 6 and 8 valves.

Shutdown Cooling System Isolation

6.a. Reactor Steam Dome Pressure—High

The Reactor Steam Dome Pressure—High Function is provided to isolate the shutdown cooling portion of the Residual Heat Removal (RHR) System. This interlock is provided only for equipment protection to prevent an intersystem LOCA scenario, and credit for the interlock is not assumed in the accident or transient analysis in the FSAR.

The Reactor Steam Dome Pressure—High signals are initiated from two switches that are connected to different taps on the RPV. Two channels of Reactor Steam Dome Pressure—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. The Function is only required to be OPERABLE in MODES 1, 2, and 3, since these are the only MODES in which the reactor can be pressurized; thus, equipment protection is needed. The Allowable Value was chosen to be low enough to protect the system equipment from overpressurization.

This Function isolates Group 2 RHR SDC isolation valves.

(continued)



BASES (continued)

ACTIONS

A Note has been provided to modify the ACTIONS related to primary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable primary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable primary containment isolation instrumentation channel.

A.1 and A.2

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours for Functions 2.a, 2.b, and 5.h; 24 hours for Functions other than Functions 1.d, 2.a, 2.b, and 5.h; and 30 days for Function 1.d has been shown to be acceptable (Refs. 5 and 6) to permit restoration of any inoperable channel to OPERABLE status. Required Actions A.1 and A.2 are modified by Notes that specify the Applicability of the Required Actions for Function 1.d when 15 of 16 channels are OPERABLE. Required Action A.2 provides an allowable out of service time of 30 days for Function 1.d when 15 of 16 channels are OPERABLE. This has been shown to be acceptable (Ref. 9) to permit restoration of the one inoperable channel to OPERABLE status. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1 or A.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable

(continued)



BASES

ACTIONS

A.1 and A.2 (continued)

channel in trip would result in an isolation), Condition C must be entered and its Required Action taken.

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in redundant isolation capability being lost for the associated penetration flow path(s). For MSL, Primary Containment, HPCI, RCIC, RWCU and SDC Isolation Functions where actuation of both trip systems is needed to isolate a penetration, the Functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip). This ensures that both trip systems will generate a trip signal from the given Function on a valid signal. For those Primary Containment, HPCI, RCIC, RWCU, and SDC isolation functions, where actuation of one trip system is needed to isolate a penetration, the Functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip, such that one trip system will generate a trip signal from the given Function on a valid signal. This ensures that at least one of the PCIVs in the associated penetration flow path can receive an isolation signal from the given Function. For all Functions except 1.c, 1.d, 3.a, 4.a, 5.a through 5.g, and 6.a, this would require both trip systems to have one channel OPERABLE or in trip. For Function 1.c, this would require both trip systems to have one channel, associated with each MSL, OPERABLE or in trip. For Function 1.d, which consists of channels that monitor several locations within a given area (e.g., different locations within the main steam tunnel area), this would require both trip systems to have one channel per location OPERABLE or in trip. For Functions 3.a, 4.a, and 6.a, this would require one trip system to have one channel OPERABLE or in trip. For Functions 5.a through 5.f, this would require both trip systems to have one channel, associated with each area, OPERABLE or in trip.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes

(continued)



BASES

ACTION

B.1 (continued)

risk while allowing time for restoration or tripping of channels.

The second Completion Time for Function 1.d when normal ventilation is not available is provided to allow the plant to avoid an MSL isolation transient when recovering from a temporary loss of ventilation in the MSL tunnel area (e.g., during performance of the secondary containment leak rate tests). As allowed by LCO 3.0.2 (and discussed in the Bases for LCO 3.0.2), the plant may intentionally enter this condition to avoid an MSL isolation transient and bypass the high temperature channels during restoration of ventilation flow. However, during the period that multiple Main Steam Tunnel Temperature - High Function channels are inoperable due to this intentional action, an additional compensatory measure is deemed necessary and shall be taken: an operator shall observe control room indications of the affected space temperatures for indications of small steam leaks. In the event of rapid increases in temperature (indicative of a steam line break), the operator shall promptly close the MSIVs. The 4 hour Completion Time is acceptable because along with the compensatory measures described above it minimizes risk while allowing time for restoration or tripping of channels.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.6.1-1. The applicable Condition specified in Table 3.3.6.1-1 is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A or B and the associated Completion Time has expired, Condition C will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

D.1, D.2.1, and D.2.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must

(continued).



BASES

ACTIONS

D.1, D.2.1, and D.2.2 (continued)

be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours (Required Actions D.2.1 and D.2.2). Alternately, the associated MSLs may be isolated (Required Action D.1), and, if allowed (i.e., plant safety analysis allows operation with an MSL isolated), operation with that MSL isolated may continue. Isolating the affected MSL accomplishes the safety function of the inoperable channel. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 2 within 6 hours.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

F.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, plant operations may continue if the affected penetration flow path(s) is isolated. Isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable channels.

For the RWCU Area Temperature-High Functions, the affected penetration flow path(s) may be considered isolated by isolating only that portion of the system in the associated room monitored by the inoperable channel. That is, if the RWCU pump room A area channel is inoperable, the pump room A area can be isolated while allowing continued RWCU operation utilizing the B RWCU pump.

(continued)



BASES

ACTIONS

F.1 (continued)

Alternately, if it is not desired to isolate the affected penetration flow path(s) (e.g., as in the case where isolating the penetration flow path(s) could result in a reactor scram), Condition G must be entered and its Required Actions taken.

The 1 hour Completion Time is acceptable because it minimizes risk while allowing sufficient time for plant operations personnel to isolate the affected penetration flow path(s).

G.1 and G.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, or any Required Action of Condition F is not met and the associated Completion Time has expired, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

H.1 and H.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the SLC System is declared inoperable or the RWCU System is isolated. Since this Function is required to ensure that the SLC System performs its intended function, sufficient remedial measures are provided by declaring the SLC System inoperable or isolating the RWCU System.

The 1 hour Completion Time is acceptable because it minimizes risk while allowing sufficient time for personnel to isolate the RWCU System.

(continued)



BASES (continued)

SURVEILLANCE
REQUIREMENTS

As noted (Note 1) at the beginning of the SRs, the SRs for each Primary Containment Isolation instrumentation Function are found in the SRs column of Table 3.3.6.1-1.

The Surveillances are modified by a Note (Note 2) to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 5 and 6) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the PCIVs will isolate the penetration flow path(s) when necessary.

SR 3.3.6.1.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.6.1.1 (continued)

channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.6.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day Frequency of SR 3.3.6.1.2 is based on the reliability analysis described in References 5 and 6.

SR 3.3.6.1.3, SR 3.3.6.1.4 and SR 3.3.6.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequencies of SR 3.3.6.1.3, SR 3.3.6.1.4, and SR 3.3.6.1.5 are based on the magnitude of equipment drift in the setpoint analysis.

SR 3.3.6.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on PCIVs in LCO 3.6.1.3 overlaps this Surveillance to provide complete testing of the assumed safety function. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.6.1.6 (continued)

Operating experience has shown these components usually pass the Surveillance when performed at the Frequency provided.

REFERENCES

1. FSAR, Section 6.5.
 2. FSAR, Chapter 14.
 3. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.
 4. FSAR, Section 4.9.3.
 5. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.
 6. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.
 7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
 8. FSAR, Section 5.2.
 9. NRC letter from Richard J. Clark to Hugh G. Parris dated August 9, 1984, Safety Evaluation for Amendment Nos. 107, 101, and 74 to Facility Operating License Nos. DPR-33, DPR-52, and DPR-68 for Browns Ferry Nuclear Plant Units 1, 2, and 3 respectively.
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B 3.3 INSTRUMENTATION

B 3.3.6.2 Secondary Containment Isolation Instrumentation

BASES

BACKGROUND

The secondary containment isolation instrumentation automatically initiates closure of appropriate secondary containment isolation valves (SCIVs) and starts the Standby Gas Treatment (SGT) System. The function of these systems, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref. 1). Secondary containment isolation and establishment of vacuum with the SGT System within the assumed time limits ensures that fission products that leak from primary containment following a DBA, or are released outside primary containment, or are released during certain operations when primary containment is not required to be OPERABLE are maintained within applicable limits.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of secondary containment isolation. Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a secondary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logic are (1) reactor vessel water level, (2) drywell pressure, (3) reactor zone exhaust high radiation, and (4) refueling floor exhaust high radiation. Redundant sensor input signals from each parameter are provided for initiation of isolation. In addition, manual initiation of the logic is provided.

The output signals from the secondary containment isolation logic isolates secondary containment and starts all three SGT subsystems to provide for the necessary filtration of fission products.

(continued)



BASES (continued)

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The isolation signals generated by the secondary containment isolation instrumentation are implicitly assumed in the safety analyses of References 1 and 2 to initiate closure of valves and start the SGT System to limit offsite doses.

Refer to LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," Applicable Safety Analyses Bases for more detail of the safety analyses.

The secondary containment isolation instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 7). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the secondary containment isolation instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have the required number of OPERABLE channels with their setpoints set within the specified Allowable Values, as shown in Table 3.3.6.2-1. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Each channel must also respond within its assumed response time, where appropriate.

Allowable Values are specified for each Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions when SCIVs and the SGT System are required.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. Reactor Vessel Water Level - Low, Level 3

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite dose release. The Reactor Vessel Water Level - Low, Level 3 Function is one of the Functions assumed to be OPERABLE and capable of providing isolation and initiation signals. The isolation and initiation systems on Reactor Vessel Water Level - Low, Level 3 support actions to ensure that any offsite releases are within the limits calculated in the safety analysis (Ref. 4).

Reactor Vessel Water Level - Low, Level 3 signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low, Level 3 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1. Reactor Vessel Water Level—Low Low, Level 2
(continued)

The Reactor Vessel Water Level—Low, Level 3 Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the Reactor Coolant System (RCS); thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, this Function is not required. In addition, the Function is also required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) because the capability of isolating potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded if core damage occurs.

2. Drywell Pressure—High

High drywell pressure can indicate a break in the reactor coolant pressure boundary (RCPB). An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite dose release. The isolation on high drywell pressure supports actions to ensure that any offsite releases are within the limits calculated in the safety analysis. However, the Drywell Pressure—High Function associated with isolation is not assumed in any FSAR accident or transient analyses. It is retained for the overall redundancy and diversity of the secondary containment isolation instrumentation as required by the NRC approved licensing basis.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure—High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude performance of the isolation function.

The Allowable Value was chosen to be the same as the ECCS Drywell Pressure—High Function Allowable Value

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

2. Drywell Pressure—High (continued)

(LCO 3.3.5.1) since this is indicative of a loss of coolant accident (LOCA).

The Drywell Pressure—High Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the RCS; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This Function is not required in MODES 4 and 5 because the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES.

3, 4. Reactor Zone and Refueling Floor Exhaust
Radiation—High

High secondary containment exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident. When Exhaust Radiation—High is detected, secondary containment isolation and actuation of the SGT System are initiated to limit the release of fission products as assumed in the FSAR safety analyses (Ref. 4).

The Exhaust Radiation—High signals are initiated from radiation detectors located on the ventilation exhausts coming from each reactor zone and the common refueling zone. There are two radiation monitors for each ventilation exhaust path. There are two pairs of radiation elements which monitor the ventilation exhaust from each zone. Each pair of radiation elements provides input to one radiation monitor. Both radiation elements must provide a High signal to trip the associated radiation monitor (two-out-of-two). However, if either radiation monitor trips, a secondary containment isolation signal is initiated (one-out-of-two). Two channels (monitors) of Reactor Zone Exhaust Radiation—High Function and two channels of Refueling Floor Exhaust Radiation—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. There is only one trip system for each Function.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3. 4. Reactor Zone and Refueling Floor Exhaust
Radiation—High (continued)

The Allowable Values are chosen to provide timely detection of nuclear system process barrier leaks inside containment but are far enough above background levels to avoid spurious isolation.

The Reactor Zone and Refueling Floor Exhaust Radiation—High Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are also required to be OPERABLE during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncover or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded.

ACTIONS

A Note has been provided to modify the ACTIONS related to secondary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable secondary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable secondary containment isolation instrumentation channel.

(continued)



BASES

ACTIONS
(continued)

A.1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours for Functions 1 and 2, and 24 hours for Functions other than Functions 1 and 2, has been shown to be acceptable (Refs. 5 and 6) to permit restoration of any inoperable channel to OPERABLE status. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition C must be entered and its Required Actions taken.

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic isolation capability for the associated secondary containment penetration flow path(s) or a complete loss of automatic initiation capability for the SGT System. A Function is considered to be maintaining secondary containment isolation capability when sufficient channels are OPERABLE or in trip, such that one trip system will generate a trip signal from the given Function on a valid signal. This ensures that one of the two SCIVs in the associated penetration flow path and two SGT subsystems can be initiated on an isolation signal from the given Function. For Functions with two one-out-of-two logic trip systems (Functions 1 and 2), this would require one trip system to have one channel OPERABLE or in trip. For Functions with one one-out-of-two logic trip system (Functions 3 and 4), this would require the trip system to have one channel OPERABLE or in trip.

(continued)



BASES

ACTIONS

B.1 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

C.1.1, C.1.2, C.2.1, and C.2.2

If any Required Action and associated Completion Time of Condition A or B are not met, the ability to isolate the secondary containment and start the SGT System cannot be ensured. Therefore, further actions must be performed to ensure the ability to maintain the secondary containment function. Isolating the associated zone (closing the ventilation supply and exhaust automatic isolation dampers) and starting the associated SGT subsystem (Required Actions C.1.1 and C.2.1) performs the intended function of the instrumentation and allows operation to continue.

Alternately, declaring the associated SCIVs or SGT subsystem(s) inoperable (Required Actions C.1.2 and C.2.2) is also acceptable since the Required Actions of the respective LCOs (LCO 3.6.4.2 and LCO 3.6.4.3) provide appropriate actions for the inoperable components.

One hour is sufficient for plant operations personnel to establish required plant conditions or to declare the associated components inoperable without unnecessarily challenging plant systems.

SURVEILLANCE
REQUIREMENTS

As noted (Note 1) at the beginning of the SRs, the SRs for each Secondary Containment Isolation instrumentation Function are located in the SRs column of Table 3.3.6.2-1.

The Surveillances are modified by a Note (Note 2) to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains secondary containment isolation capability. Upon completion of the Surveillance, or expiration of the 6 hour

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 5 and 6) assumption of the average time required to perform channel surveillance. That analysis demonstrated the 6 hour testing allowance does not significantly reduce the probability that the SCIVs will isolate the associated penetration flow paths and that the SGT System will initiate when necessary.

The Surveillances are modified by a third Note (Note 3) to indicate that for Functions 2.c and 2.d, when a channel is placed in an inoperable status solely for performance of required testing or maintenance, entry into associated Conditions and Required Actions may be delayed for up to 6 hours for a CHANNEL FUNCTIONAL TEST and for up to 24 hours for a CHANNEL CALIBRATION or maintenance, provided the downscale trip of the inoperable channel is placed in the tripped condition. Upon completion of the Surveillance or maintenance, or expiration of the 6 hour or 24 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

SR 3.3.6.2.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.6.2.1 (continued)

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel status during normal operational use of the displays associated with channels required by the LCO.

SR 3.3.6.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of References 5 and 6.

SR 3.3.6.2.3 and SR 3.3.6.2.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on SCIVs and the SGT System in LCO 3.6.4.2 and LCO 3.6.4.3, respectively, overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency for Functions 1 and 2 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 184 day Frequency for Functions 3 and 4 is based on operating experience and equipment capability.

Operating experience has shown that these components usually pass the Surveillance when performed at these Frequencies. Therefore, the Frequencies were found to be acceptable from a reliability standpoint.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.6.2.4 (continued)

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of SR 3.3.6.2.4 is based on the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. FSAR, Chapter 5 and Section 7.3.5.
 2. FSAR, Chapter 14.
 3. FSAR, Section 14.6.3.5.
 4. FSAR, Sections 14.6.3.6 and 14.6.4.5.
 5. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.
 6. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.
 7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.3 INSTRUMENTATION

B 3.3.7.1 Control Room Emergency Ventilation (CREV) System Instrumentation

BASES

BACKGROUND

The CREV System is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. Two independent CREV subsystems are each capable of fulfilling the stated safety function. The instrumentation and controls for the CREV System automatically initiate action to pressurize the control room (CR) to minimize the consequences of radioactive material in the control room environment.

In the event of a Reactor Vessel Water Level-Low, Level 3, Drywell Pressure-High, Reactor Zone Exhaust Radiation-High, Refueling Floor Exhaust Radiation-High, or Control Room Air Supply Duct Radiation-High signal, the CREV System is automatically started in the pressurization mode. The air is then recirculated through the charcoal filter, and sufficient outside air is drawn in through the normal intake to maintain the CR slightly pressurized.

The CREV System instrumentation has one or two trip systems, which can initiate both CREV subsystems (only the selected subsystem will be initiated) (Ref. 1). Each trip system receives input from each of the Functions listed above. The Functions are arranged as follows for each trip system. The Reactor Vessel Water Level-Low, Level 3 and Drywell Pressure-High are each arranged in a one-out-of-two taken twice logic (these signals are the same that isolate the primary containment). The Reactor Zone Exhaust Radiation-High, Refueling Floor Exhaust Radiation-High and Control Room Air Supply Duct Radiation-High (only one trip system) are each arranged in a one-out-of-two logic. The channels include electronic equipment (e.g., trip relays) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a CREV System initiation signal to the initiation logic.

(continued)



BASES (continued)

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The ability of the CREV System to maintain the habitability of the CR is explicitly assumed for certain accidents as discussed in the FSAR safety analyses (Ref. 2). CREV System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents, does not exceed the limits set by GDC 19 of 10 CFR 50, Appendix A.

CREV System instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 5).

The OPERABILITY of the CREV System instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.7.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint).

Allowable Values are specified for each CREV System Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip relay) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. Reactor Vessel Water Level - Low, Level 3

Low reactor pressure vessel (RPV) water level indicates that the capability of cooling the fuel may be threatened. A low reactor vessel water level could indicate a LOCA and will automatically initiate the CREV System, since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

Reactor Vessel Water Level - Low, Level 3 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low, Level 3 Function are available (two channels per trip system) and are required to be OPERABLE to ensure that a single instrument failure cannot preclude CREV System initiation. The Reactor Vessel Water Level - Low, Level 3 instrumentation which provides input signals to the CREV System initiation logic is the same instrumentation which provides the input signals for the Primary Containment Isolation System logic (LCO 3.3.6.1).

The Reactor Vessel Water Level - Low, Level 3 Function is required to be OPERABLE in MODES 1, 2, and 3, and during operations with a potential for draining the reactor vessel (OPDRVs) to ensure that the control room personnel are protected during a LOCA. In MODES 4 and 5 at times other than OPDRVs, the probability of a vessel draindown event resulting in a release of radioactive material into the environment is minimal. In addition, adequate protection is performed by the Control Room Air Supply Duct Radiation - High Function. Therefore, this Function is not required in other MODES and specified conditions.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

2. Drywell Pressure—High

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary. A high drywell pressure signal could indicate a LOCA and will automatically initiate the CREV System, since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

Drywell Pressure—High signals are initiated from four pressure transmitters that sense drywell pressure. Four channels of Drywell Pressure—High Function are available (two channels per trip system) and are required to be OPERABLE to ensure that no single instrument failure can preclude CREV System initiation. The Drywell Pressure—High Allowable Value was chosen to be the same as the ECCS Drywell Pressure—High Allowable Value (LCO 3.3.5.1).

The Drywell Pressure—High Function is required to be OPERABLE in MODES 1, 2, and 3 to ensure that control room personnel are protected in the event of a LOCA. In MODES 4 and 5, the Drywell Pressure—High Function is not required since there is insufficient energy in the reactor to pressurize the drywell to the Drywell Pressure—High setpoint.

3., 4. Reactor Zone and Refueling Floor Exhaust
Radiation—High

High secondary containment exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB. When Exhaust Radiation—High is detected, valves whose penetrations communicate with the primary containment atmosphere are isolated to limit the release of fission products. Additionally, high radiation in the refueling floor exhaust could be the result of a fuel handling accident. A reactor zone or refueling floor exhaust high radiation signal will automatically initiate the CREV System, since this radiation release could result in radiation exposure to control room personnel.

The reactor zone and refueling floor exhaust radiation equipment consists of two independent monitors and channels

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3., 4. Reactor Zone and Refueling Floor Exhaust
Radiation—High (continued)

located on the ventilation exhaust piping coming from the reactor building and the refueling zones, respectively. Two channels of each function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude CREV System initiation. There is only one trip system for each Function. The Allowable Value was selected to ensure that the Function will promptly detect high activity that could threaten exposure to control room personnel.

The Reactor Zone and Refueling Floor Exhaust Radiation—High Functions are required to be OPERABLE in MODES 1, 2, and 3 and during movement of irradiated fuel assemblies in the secondary containment, CORE ALTERATIONS, and operations with a potential for draining the reactor vessel (OPDRVs), to ensure that control room personnel are protected during a LOCA, fuel handling event, or vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., CORE ALTERATIONS), the probability of a LOCA or fuel damage is low; thus, the Function is not required.

5. Control Room Air Supply Duct Radiation—High

The control room air supply duct radiation monitors measure radiation levels exterior to the inlet ducting of the CR. A high radiation level may pose a threat to CR personnel; thus, the CREV System is automatically initiated on a control room air supply duct high radiation signal.

The Control Room Air Supply Duct Radiation—High Function consists of two independent monitors. Two channels of Control Room Air Supply Duct Radiation—High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude CREV System initiation. There is only one trip system for this Function. The Allowable Value was selected to ensure protection of the control room personnel.

The Control Room Air Supply Duct Radiation—High Function is required to be OPERABLE in MODES 1, 2, and 3 and during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

5. Control Room Air Supply Duct Radiation — High
(continued)

assemblies in the secondary containment, to ensure that control room personnel are protected during a LOCA, fuel handling event, or vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., CORE ALTERATIONS), the probability of a LOCA or fuel damage is low; thus, the Function is not required.

ACTIONS

A Note has been provided to modify the ACTIONS related to CREV System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable CREV System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable CREV System instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.7.1-1. The applicable Condition specified in the Table is Function dependent. Each time a channel is discovered inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

B.1 and B.2

Because of the diversity of sensors available to provide initiation signals and the redundancy of the CREV System design, an allowable out of service time of 12 hours has been shown to be acceptable (Refs. 3 and 4) to permit restoration of any inoperable channel to OPERABLE status.

(continued)



BASES

ACTIONS

B.1 and B.2 (continued)

However, this out of service time is only acceptable provided the associated Function is still maintaining CREV System initiation capability. A Function is considered to be maintaining CREV System initiation capability when sufficient channels are OPERABLE or in trip such that one trip system will generate an initiation signal from the given Function on a valid signal. In this situation (loss of CREV System initiation capability), the 12 hour allowance of Required Action B.2 is not appropriate. If the Function is not maintaining CREV System initiation capability, the CREV System must be declared inoperable within 1 hour of discovery of the loss of CREV System initiation capability in both trip systems.

The 1 hour Completion Time (B.1) is acceptable because it minimizes risk while allowing time for restoring or tripping of channels.

If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition E must be entered and its Required Action taken.

C.1 and C.2

Because of the diversity of sensors available to provide initiation signals and the redundancy of the CREV System design, an allowable out of service time of 24 hours is provided to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function is still maintaining CREV System initiation capability. In this situation (loss of CREV System initiation capability), the 24 hour allowance of Required Action C.2 is not appropriate. If the Function is not maintaining CREV System initiation capability, the CREV System must be declared inoperable

(continued)



BASES

ACTIONS

C.1 and C.2 (continued)

within 1 hour of discovery of the loss of CREV System initiation capability in both trip systems.

The 1 hour Completion Time (C.1) is acceptable because it minimizes risk while allowing time for restoring or tripping of channels.

If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action C.2. Placing the inoperable channel in trip performs the intended function of the channel (starts the selected CREV subsystem in the pressurization mode). Alternately, if it is not desired to place the channel in trip (e.g., as in the case where it is not desired to start the subsystem), Condition E must be entered and its Required Action taken.

D.1, D.2, and D.3

Because of the diversity of sensors available to provide initiation signals and the redundancy of the CREV System design, Required Action D.1 allows continued operation with an inoperable channel provided repair is initiated in a timely manner and the remaining OPERABLE channel is functionally tested once per 24 hours. With two channels of the Control Room Air Supply Duct Radiation - High function inoperable (Required Actions D.2 and D.3), an allowed outage time of 30 days is provided to restore at least one channel to OPERABLE status provided that the alternate monitoring capability is verified functional once per 12 hours. The alternate monitoring capability is provided by the control room particulate monitor (RM-90-53) and radiation monitor (RE-90-8). These monitors alarm in the control room on high activity. Upon receipt of these alarms, the operator is required to manually isolate the control room and manually initiate the emergency pressurization system. The 30 day allowed outage time is based on verifying functional capability of these two monitors and the administrative controls that require operator action to manually initiate a CREV subsystem.

(continued)



BASES

ACTIONS
(continued)

E.1 and E.2

With any Required Action and associated Completion Time not met, the associated CREV subsystem(s) must be placed in the pressurization mode of operation per Required Action E.1 to ensure that control room personnel will be protected in the event of a Design Basis Accident. The method used to place the CREV subsystem(s) in operation must provide for automatically re-initiating the subsystem(s) upon restoration of power following a loss of power to the CREV subsystem(s). Alternately, if it is not desired to start the subsystem(s), the CREV subsystem(s) associated with inoperable, untripped channels must be declared inoperable within 1 hour.

The 1-hour Completion Time is intended to allow the operator time to place the CREV subsystem(s) in operation. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels, for placing the associated CREV subsystem(s) in operation, or for entering the applicable Conditions and Required Actions for the inoperable CREV subsystem(s).

SURVEILLANCE
REQUIREMENTS

As noted (Note 1) at the beginning of the SRs, the SRs for each CREV System instrumentation Function are located in the SRs column of Table 3.3.7.1-1.

The Surveillances are modified by a Note (Note 2) to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains CREV System initiation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 3 and 4) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the CREV System will initiate when necessary.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

The Surveillances are modified by a third Note (Note 3) to indicate that for Functions 3 and 4, when a channel is placed in an inoperable status solely for performance of required testing or maintenance, entry into associated Conditions and Required Actions may be delayed for up to 6 hours for a CHANNEL FUNCTIONAL TEST and for up to 24 hours for a CHANNEL CALIBRATION or maintenance, provided the downscale trip of the inoperable channel is placed in the tripped condition. Upon completion of the Surveillance or maintenance, or expiration of the 6 hour or 24 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

SR 3.3.7.1.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel status during normal operational use of the displays associated with channels required by the LCO.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.7.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analyses of References 3 and 4.

SR 3.3.7.1.3 and SR 3.3.7.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequencies are based upon the magnitude of equipment drift in the setpoint analysis.

SR 3.3.7.1.4 and SR 3.3.7.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.7.3, "Control Room Emergency Ventilation (CREV) System," overlaps this Surveillance to provide complete testing of the assumed safety function.

The 184 day Frequency for Function 5 is based on equipment capability. The 18 month Frequency for Functions 1, 2, 3, and 4 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at their designated Frequencies.

(continued)



BASES (continued)

- REFERENCES
1. FSAR, Section 10.12.5.3.
 2. FSAR, Section 14.6.3.7.
 3. GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.
 4. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.
 5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.3 INSTRUMENTATION

B 3.3.8.1 Loss of Power (LOP) Instrumentation

BASES

BACKGROUND

Successful operation of the required safety functions of the Emergency Core Cooling Systems (ECCS) is dependent upon the availability of adequate power sources for energizing the various components such as pump motors, motor operated valves, and the associated control components. The LOP instrumentation monitors the 4.16 kV shutdown boards. Offsite power is the preferred source of power for the 4.16 kV shutdown boards. If the monitors determine that insufficient power is available, the boards are disconnected from the offsite power sources and connected to the onsite diesel generator (DG) power sources.

Each 4.16 kV shutdown board has its own independent LOP instrumentation and associated trip logic. The voltage for each board is monitored at two levels, which can be considered as two different undervoltage functions: Loss of Voltage and 4.16 kV Shutdown Board Undervoltage Degraded Voltage. Each function causes various board transfers and disconnects.

The Degraded Voltage function is monitored by three undervoltage relays for each shutdown board, whose outputs are arranged in a two-out-of-three logic configuration (Ref. 1). The channels include electronic equipment (e.g., trip relays) that compare measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay deenergizes, which then outputs a LOP trip signal to the shutdown board logic.

The Loss of Voltage function is monitored by two undervoltage relay pairs for each shutdown board, where outputs are arranged in a two-out-of-two logic configuration (Ref. 1). The channels include four electro-mechanical relays, two of which must deenergize to start the associated diesel generator and another two which must deenergize to initiate load shed of the associated 4.16 kV shutdown board.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The LOP instrumentation is required for Engineered Safety Features to function in any accident with a loss of offsite power. The required channels of LOP instrumentation ensure that the ECCS and other assumed systems powered from the DGs, provide plant protection in the event of any of the Reference 2, 3, and 4 analyzed accidents in which a loss of offsite power is assumed. The initiation of the DGs on loss of offsite power, and subsequent initiation of the ECCS, ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Accident analyses credit the loading of the DG based on the loss of offsite power concurrent with a loss of coolant accident. The diesel starting and loading times have been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power.

The LOP instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 5).

The OPERABILITY of the LOP instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.8.1-1. Each Function must have a required number of OPERABLE channels per 4.16 kV shutdown board, with their setpoints within the specified Allowable Values. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

The Allowable Values are specified for each Function in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within the Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., degraded voltage), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip relay) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the

(continued)



BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for unit channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. 4.16 kV Shutdown Board Undervoltage (Loss of Voltage)

Loss of voltage on a 4.16 kV shutdown board indicates that offsite power may be completely lost to the respective shutdown board and is unable to supply sufficient power for proper operation of the applicable equipment. Therefore, the power supply to the board is transferred from offsite power to DG power upon total loss of shutdown board voltage for 1.5 seconds. The transfer will not occur if the voltage recovers to the specified Allowable Value for Reset Voltage within 1.5 seconds. This ensures that adequate power will be available to the required equipment.

The Time Delay Allowable Values are long enough to provide time for the offsite power supply to recover to normal voltages, but short enough to ensure that power is available to the required equipment.

One channel of 4.16 kV Shutdown Board Undervoltage (Loss of Voltage) Function per associated shutdown board is only required to be OPERABLE when the associated DG is required to be OPERABLE to ensure that no single instrument failure can preclude the DG function. Refer to LCO 3.8.1, "AC Sources - Operating," and 3.8.2, "AC Sources - Shutdown," for Applicability Bases for the DGs.

2. 4.16 kV Shutdown Board Undervoltage (Degraded Voltage)

A reduced voltage condition on a 4.16 kV shutdown board indicates that, while offsite power may not be completely

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
LCO, and
APPLICABILITY

2. 4.16 kV Shutdown Board Undervoltage (Degraded Voltage)
(continued)

lost to the respective shutdown board, available power maybe insufficient for starting large ECCS motors without risking damage to the motors that could disable the ECCS function. Therefore, power supply to the board is transferred from offsite power to onsite DG power when the voltage on the board drops below the Degraded Voltage Function Allowable Values (degraded voltage with a time delay). This ensures that adequate power will be available to the required equipment.

The Board Undervoltage Allowable Values are low enough to prevent inadvertent power supply transfer, but high enough to ensure that sufficient power is available to the required equipment. The Time Delay Allowable Values are long enough to provide time for the offsite power supply to recover to normal voltages, but short enough to ensure that sufficient power is available to the required equipment.

One channel of 4.16 kV Shutdown Board Undervoltage (Degraded Voltage) Function per associated board is only required to be OPERABLE when the associated DG is required to be OPERABLE to ensure that no single instrument failure can preclude the DG function. Refer to LCO 3.8.1 and LCO 3.8.2 for Applicability Bases for the DGs.

ACTIONS

A Note has been provided to modify the ACTIONS related to LOP instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable LOP instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable LOP instrumentation channel.

(continued)



BASES

ACTIONS
(continued)

A.1.

With one of the three phase-to-phase degraded voltage relays inoperable, Required Action A.1 provides a 15 day allowable out of service time to restore the relay to OPERABLE status. The 15 day allowable out of service time is justified based on the two-out-of-three permissive logic scheme provided for these relays. If the inoperable relay cannot be restored to OPERABLE status within the allowable out of service time, the degraded voltage relay channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

B.1

With one or more loss of voltage relay channels inoperable, the Function is not capable of performing the intended function. Required Action B.1 provides a 10 day allowable out of service time since the degraded voltage relay channel on the same shutdown board is independent of the loss of voltage relay channel and will continue to function and start the diesel generators on a complete loss of voltage. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

(continued)



BASES

ACTIONS
(continued)

C.1

With one or more degraded voltage relay channels inoperable, the Function is not capable of performing the intended function. Required Action C.1 provides a 10 day allowable out of service time, since the loss of voltage relay channel on the same shutdown board is independent of the degraded voltage relay channel and will continue to function and start the diesel generators on a complete loss of voltage. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action C.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

D.1 and D.2

With the degraded voltage relay channel and the loss of voltage relay channel inoperable on the same shutdown board, the associated diesel generator will not automatically start upon degraded voltage or complete loss of voltage on that shutdown board. In this situation, Required Action D.2 provides a 5 day allowable out of service time provided the other shutdown boards and undervoltage relays are OPERABLE. Immediate verification of the OPERABILITY of the other shutdown boards and undervoltage relays is therefore required (Required Action D.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this equipment cannot be verified, however, Condition E must be entered immediately. The 5 day allowable out of service time is justified based on the remaining redundancy of the 4.16 kV Shutdown Boards. The 4.16 kV Shutdown Boards have a similar allowable out of service time. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per

(continued)



BASES

ACTIONS

D.1 and D.2 (continued)

Required Action D.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

E.1

If any Required Action and associated Completion Time are not met, the associated Function is not capable of performing the intended function. Therefore, the associated DG(s) is declared inoperable immediately. This requires entry into applicable Conditions and Required Actions of LCO 3.8.1 and LCO 3.8.2, which provide appropriate actions for the inoperable DG(s).

SURVEILLANCE
REQUIREMENTS

As noted (Note 1) at the beginning of the SRs, the SRs for each LOP instrumentation Function are located in the SRs column of Table 3.3.8.1-1.

The Surveillances are modified by a Note (Note 2) to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 2 hours provided the associated Function maintains initiation capability for three DGs. The loss of function for one DG for this short period is appropriate since only three of four DGs are required to start within the required times and because there is not appreciable impact on risk. Upon completion of the Surveillance, or expiration of the 2 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.8.1.1 and SR 3.3.8.1.2

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency is based upon the calibration interval assumed in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.8.1.3

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The system functional testing performed in LCO 3.8.1 and LCO 3.8.2 overlaps this Surveillance to provide complete testing of the assumed safety functions.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. FSAR, Figure 8.4-4.
 2. FSAR, Section 6.5.
 3. FSAR, Section 8.5.4.
 4. FSAR, Chapter 14.
 5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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B 3.3 INSTRUMENTATION

B 3.3.8.2 Reactor Protection System (RPS) Electric Power Monitoring

BASES

BACKGROUND

RPS Electric Power Monitoring System is provided to isolate the RPS bus from the motor generator (MG) set or an alternate power supply in the event of overvoltage, undervoltage, or underfrequency. This system protects the loads connected to the RPS bus against unacceptable voltage and frequency conditions (Ref. 1) and forms an important part of the primary success path of the essential safety circuits. Some of the essential equipment powered from the RPS buses includes the RPS logic and scram solenoids.

RPS electric power monitoring assembly will detect any abnormal high or low voltage or low frequency condition in the outputs of the two MG sets or the alternate power supply and will de-energize its respective RPS bus, thereby causing all safety functions normally powered by this bus to de-energize.

In the event of failure of an RPS Electric Power Monitoring System (e.g., both in series electric power monitoring assemblies), the RPS loads may experience significant effects from the unmonitored power supply. Deviation from the nominal conditions can potentially cause damage to the scram solenoids and other Class 1E devices.

In the event of a low voltage condition for an extended period of time, the scram solenoids can chatter and potentially lose their pneumatic control capability, resulting in a loss of primary scram action.

In the event of an overvoltage condition, the RPS logic relays and scram solenoids may experience a voltage higher than their design voltage. If the overvoltage condition persists for an extended time period, it may cause equipment degradation and the loss of plant safety function.

Two redundant Class 1E contactors are connected in series between each RPS bus and its MG set, and between each RPS bus and its alternate power supply. Each of these contactors has an associated independent set of Class 1E overvoltage, undervoltage, and underfrequency sensing logic. Together, a contactor and its sensing logic constitute an

(continued)



BASES

BACKGROUND
(continued)

electric power monitoring assembly. If the output of the MG set exceeds predetermined limits of overvoltage, undervoltage, or underfrequency, for > 4 seconds, a trip relay driven by this logic circuitry opens the contactor, which removes the associated power supply from service. The timer is common to the three trip relays.

APPLICABLE
SAFETY ANALYSES

The RPS electric power monitoring is necessary to meet the assumptions of the safety analyses by ensuring that the equipment powered from the RPS buses can perform its intended function. RPS electric power monitoring provides protection to the RPS and other systems that receive power from the RPS buses, by acting to disconnect the RPS from the power supply under specified conditions that could damage the RPS bus powered equipment.

RPS electric power monitoring satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).

LCO

The OPERABILITY of each RPS electric power monitoring assembly is dependent on the OPERABILITY of the overvoltage, undervoltage, and underfrequency logic, as well as the OPERABILITY of the associated contactor. Two electric power monitoring assemblies are required to be OPERABLE for each inservice power supply. This provides redundant protection against any abnormal voltage or frequency conditions to ensure that no single RPS electric power monitoring assembly failure can preclude the function of RPS bus powered components. Each inservice electric power monitoring assembly's trip logic setpoints are required to be within the specified Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint procedures (nominal trip setpoint).

Allowable Values are specified for each RPS electric power monitoring assembly trip logic (refer to SR 3.3.8.2.2). Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected based on engineering judgment and operational experience to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its

(continued)



BASES

LCO
(continued)

Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., overvoltage), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip relay) changes state.

The Allowable Values for the instrument settings are based on the RPS continuously providing ≥ 56 Hz, $120\text{ V} \pm 10\%$ (to all equipment), and $115\text{ V} \pm 10\text{ V}$ (to scram and MSIV solenoids). The most limiting voltage requirement and associated line losses determine the settings of the electric power monitoring instrument channels. The settings are calculated based on the loads on the buses and RPS MG set or alternate power supply being 120 VAC and 60 Hz.

APPLICABILITY

The operation of the RPS electric power monitoring assemblies is essential to disconnect the RPS bus powered components from the MG set or alternate power supply during abnormal voltage or frequency conditions. Since the degradation of a nonclass 1E source supplying power to the RPS bus can occur as a result of any random single failure, the OPERABILITY of the RPS electric power monitoring assemblies is required when the RPS bus powered components are required to be OPERABLE. This results in the RPS Electric Power Monitoring System OPERABILITY being required in MODES 1, 2, and 3; and in MODES 4 and 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies (a control rod withdrawn in MODE 4 is only allowed by Special Operations LCO 3.10.4, "Single Control Rod Withdrawal - Cold Shutdown").

ACTIONS

A.1

If one RPS electric power monitoring assembly for an inservice power supply (MG set or alternate) is inoperable, or one RPS electric power monitoring assembly on each inservice power supply is inoperable, the OPERABLE assembly

(continued)



BASES

ACTIONS

A.1 (continued)

will still provide protection to the RPS bus powered components under degraded voltage or frequency conditions. However, the reliability and redundancy of the RPS Electric Power Monitoring System is reduced, and only a limited time (72 hours) is allowed to restore the inoperable assembly to OPERABLE status. If the inoperable assembly cannot be restored to OPERABLE status, the associated power supply(s) must be removed from service (Required Action A.1). This places the RPS bus in a safe condition. An alternate power supply with OPERABLE power monitoring assemblies may then be used to power the RPS bus.

The 72 hour Completion Time takes into account the remaining OPERABLE electric power monitoring assembly and the low probability of an event requiring RPS electric power monitoring protection occurring during this period. It allows time for plant operations personnel to take corrective actions or to place the plant in the required condition in an orderly manner and without challenging plant systems.

Alternately, if it is not desired to remove the power supply from service (e.g., as in the case where removing the power supply(s) from service would result in a scram or isolation), Condition C or D, as applicable, must be entered and its Required Actions taken.

B.1

If both power monitoring assemblies for an inservice power supply (MG set or alternate) are inoperable or both power monitoring assemblies in each inservice power supply are inoperable, the system protective function is lost. In this condition, 1 hour is allowed to restore one assembly to OPERABLE status for each inservice power supply. If one inoperable assembly for each inservice power supply cannot be restored to OPERABLE status, the associated power supply(s) must be removed from service within 1 hour (Required Action B.1). An alternate power supply with OPERABLE assemblies may then be used to power one RPS bus. The 1 hour Completion Time is sufficient for the plant operations personnel to take corrective actions and is acceptable because it minimizes risk while allowing time for

(continued)



BASES

ACTIONS

B.1 (continued)

restoration or removal from service of the electric power monitoring assemblies.

Alternately, if it is not desired to remove the power supply(s) from service (e.g., as in the case where removing the power supply(s) from service would result in a scram or isolation), Condition C or D, as applicable, must be entered and its Required Actions taken.

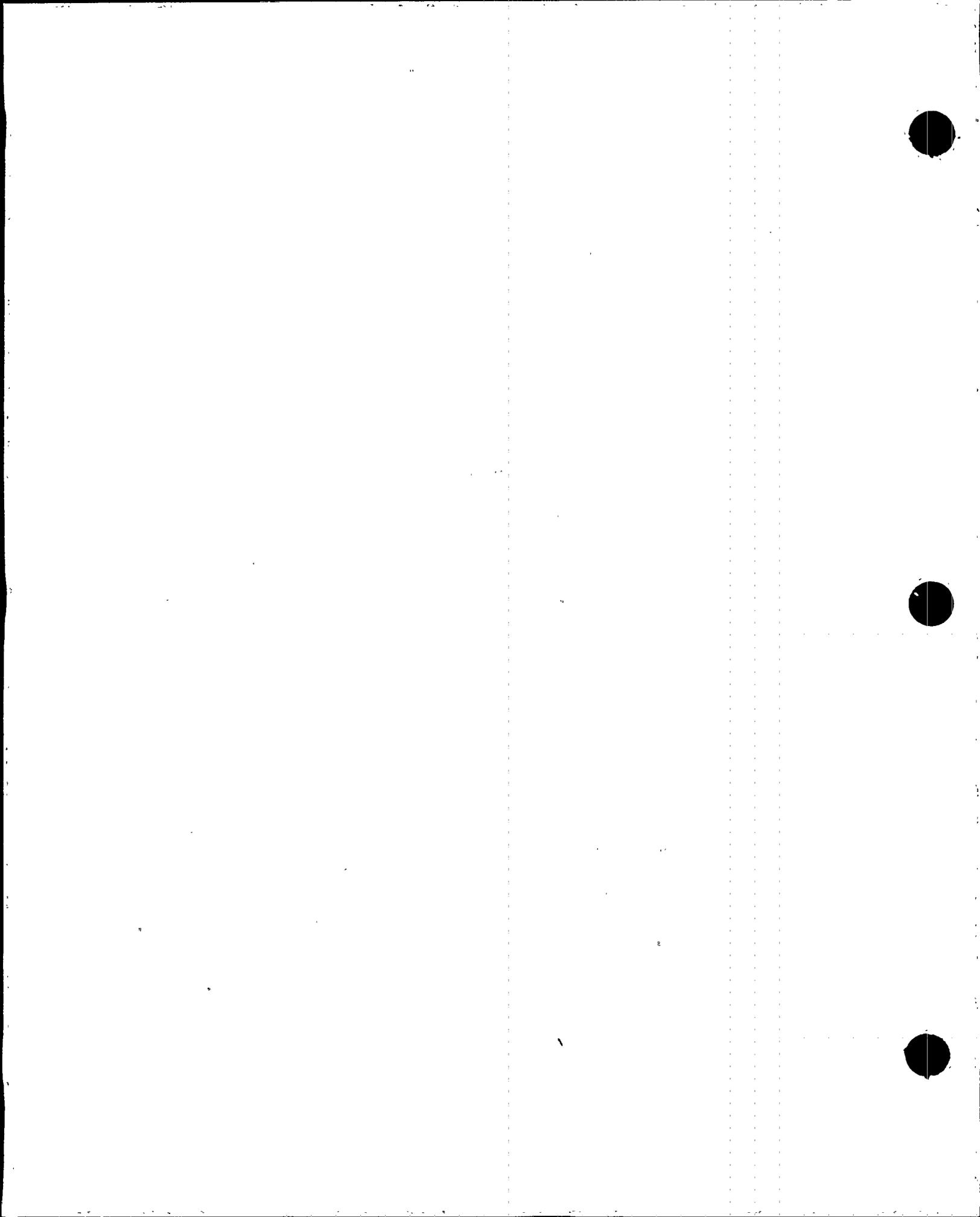
C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B are not met in MODE 1, 2, or 3, a plant shutdown must be performed. This places the plant in a condition where minimal equipment, powered through the inoperable RPS electric power monitoring assembly(s), is required and ensures that the safety function of the RPS (e.g., scram of control rods) is not required. The plant shutdown is accomplished by placing the plant in MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If any Required Action and associated Completion Time of Condition A or B are not met in MODE 4 or 5, with any control rod withdrawn from a core cell containing one or more fuel assemblies, the operator must immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Required Action D.1 results in the least reactive condition for the reactor core and ensures that the safety function of the RPS (e.g., scram of control rods) is not required.

(continued)



BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.3.8.2.1

A CHANNEL FUNCTIONAL TEST is performed on each overvoltage, undervoltage, and underfrequency channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

As noted in the Surveillance, the CHANNEL FUNCTIONAL TEST is only required to be performed while the plant is in a condition in which the loss of the RPS bus will not jeopardize steady state power operation (the design of the system is such that the power source must be removed from service to conduct the Surveillance). The 24 hours is intended to indicate an outage of sufficient duration to allow for scheduling and proper performance of the Surveillance.

The 184 day Frequency and the Note in the Surveillance are based on guidance provided in Generic Letter 91-09 (Ref. 2).

SR 3.3.8.2.2

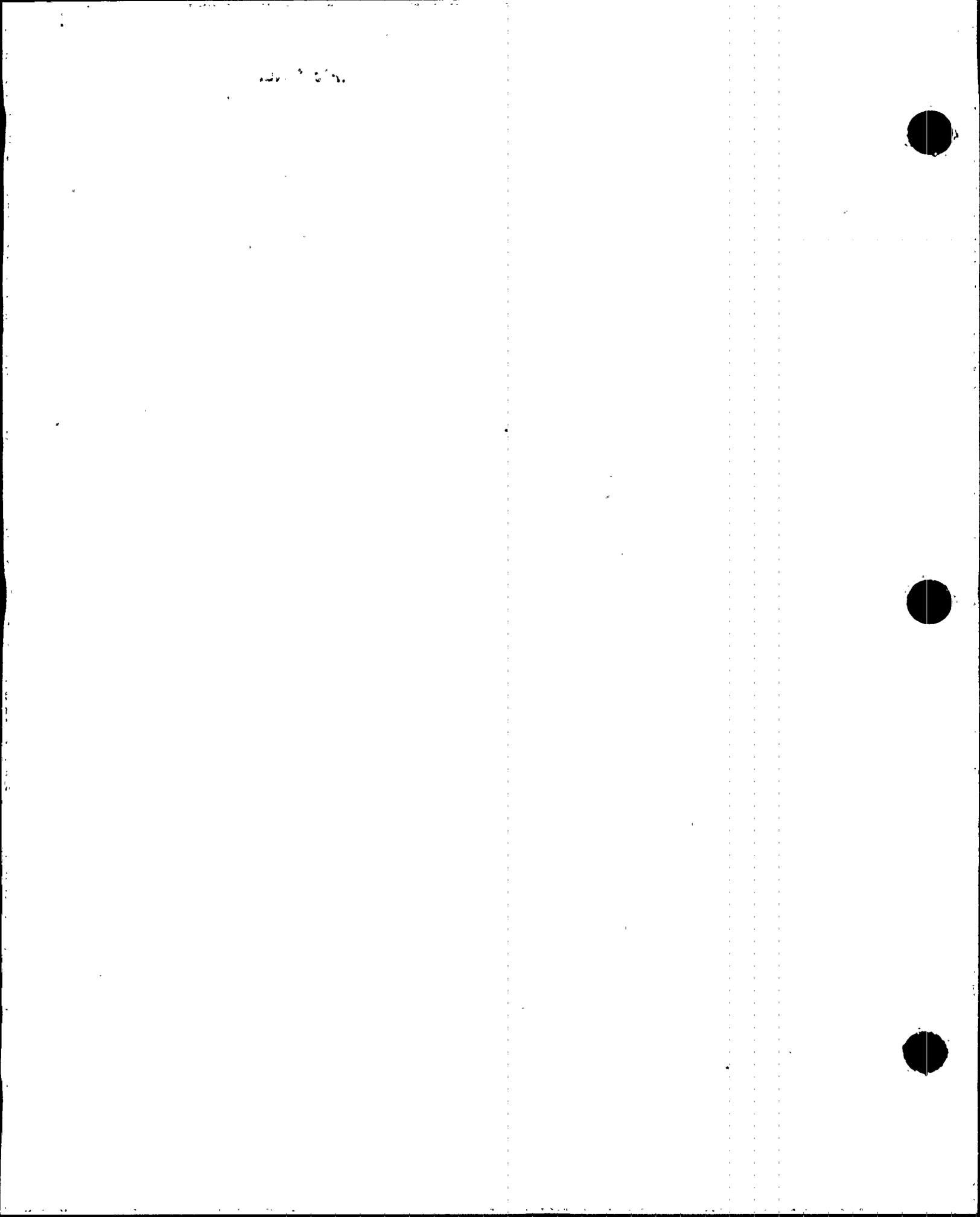
CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency is based on the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.8.2.3

Performance of a system functional test demonstrates that, with a required system actuation (simulated or actual) signal, the logic of the system will automatically trip open the associated power monitoring assembly. Only one signal per power monitoring assembly is required to be tested. This Surveillance overlaps with the CHANNEL CALIBRATION to provide complete testing of the safety function. The system

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

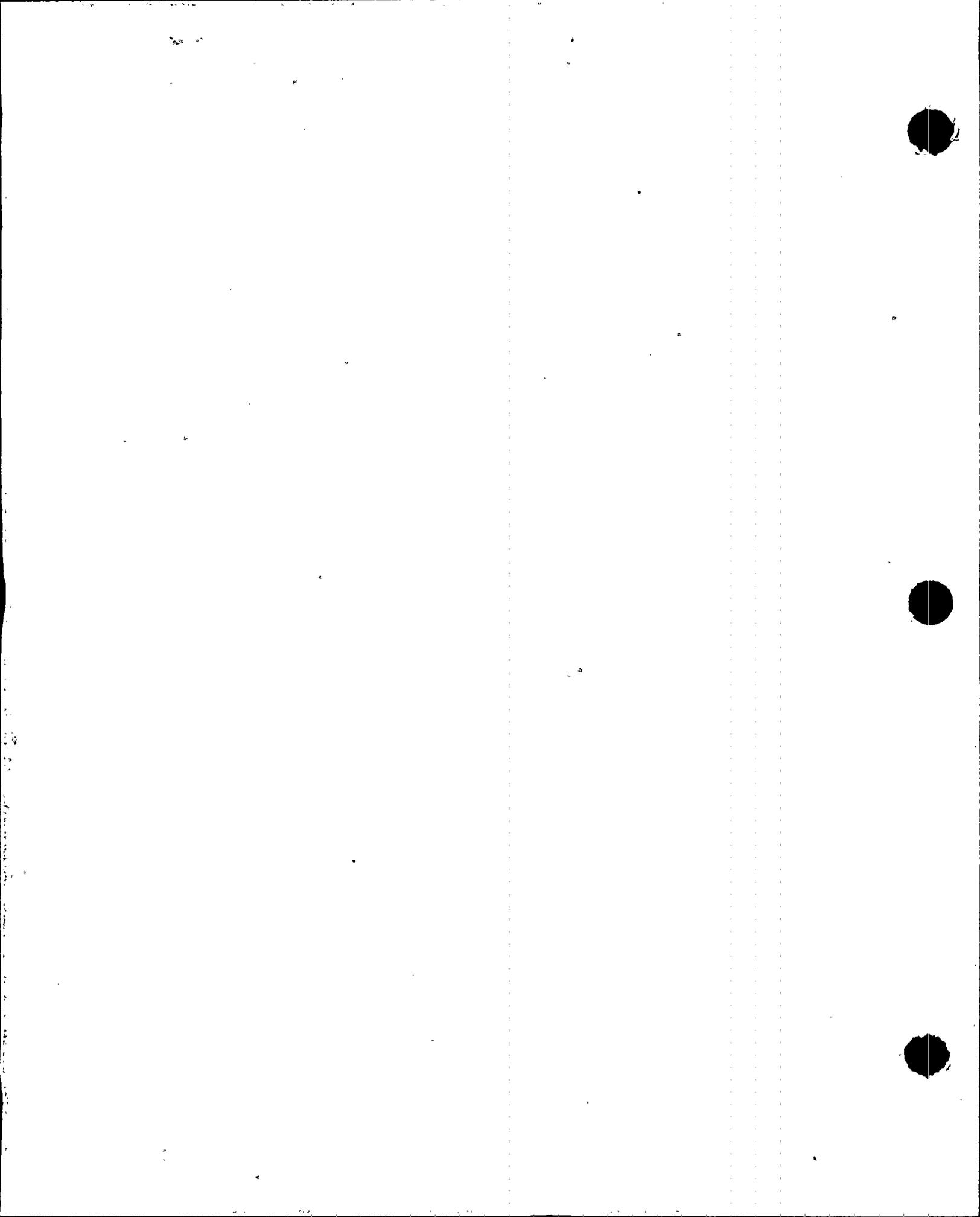
SR 3.3.8.2.3 (continued)

functional test of the Class 1E contactors is included as part of this test to provide complete testing of the safety function. If the contactors are incapable of operating, the associated electric power monitoring assembly would be inoperable.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. FSAR, Section 7.2.3.2.
 2. NRC Generic Letter 91-09, "Modification of Surveillance Interval for the Electrical Protective Assemblies in Power Supplies for the Reactor Protection System."
 3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
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BROWNS FERRY NUCLEAR PLANT

IMPROVED STANDARD TECHNICAL SPECIFICATIONS

**Enclosure V
Volume 16**

TENNESSEE VALLEY AUTHORITY

UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

(A1) 1.0 USE AND APPLICATION
1.0 DEFINITIONS

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

Specification 1.1

(A2)

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

Definitions

(A1) TERM

(A25)

A. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit requires unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

(A26)

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

(A27)

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

(A3)

1. In the event a Limiting Condition for Operation and/or associated requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours unless corrective measures are completed that permit operation under the permissible discovery or until the reactor is placed in an operational condition in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications. This provides actions to be taken for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. For example, if a specification calls for two systems (or subsystems) to be operable and provides for explicit requirements if one system (or subsystem) is inoperable, then if both systems (or subsystems) are inoperable the unit is to be in at least Hot Standby in 6 hours and in Cold Shutdown within the following 30 hours if the inoperable condition is not corrected.

(A1)

2. When a system, subsystem, train, component, or device is determined to be inoperable solely because its onsite power source is inoperable, or solely because its offsite 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 offsite or diesel 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 these requirements. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least Hot Standby within 6 hours, and in at least Cold Shutdown within the following 30 hours. This definition is not applicable in Cold Shutdown or Refueling. This provision describes what additional conditions must be satisfied to permit operation to continue consistent with the specifications for power sources, when an offsite or onsite power source is not operable. It specifically prohibits operation when one division is inoperable because its offsite or diesel power source is inoperable and a system, subsystem, train, component, or device in another division is inoperable for another reason. This provision permits the requirements associated with individual systems, subsystems, trains, components, or devices to be consistent with the requirements of the associated electrical power source. It allows operation to be governed by the time limit of the requirements associated with the Limiting Condition For Operation for the offsite or diesel power source, not the individual requirements for each system, subsystem, train, component, or device that is determined to be inoperable solely because of the inoperability of its offsite or diesel power source.

(A3)

(A28)

D. ~~PRIOR TO STARTUP~~ - Prior to withdrawing the first control rod for the purpose of making the reactor critical.

E. ~~Operable - Operability~~ - A system, subsystem, ~~train~~, component, or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition ~~shall be the assumption that all necessary attendant~~

(A5)

and

~~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).~~

(A29)

F. ~~Operating~~ - ~~Operating~~ means that a system or component is performing its intended functions in its required manner.

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

(A30)



NOV 13 1988

1.0 ~~DEFINITIONS (Cont'd)~~

H. ~~Reactor Power Operation~~ - Reactor power operation is any operation in the ~~STARTUP/HOT STANDBY~~ or ~~RUN~~ MODE with the reactor critical and above 1 percent rated power.

A6

I. ~~STARTUP CONDITION~~ - The reactor is in the ~~STARTUP CONDITION~~ when the withdrawal of control rods for the purpose of making the reactor critical has begun, reactor power is less than or equal to 1 percent of rated, and the reactor is in the ~~STARTUP/HOT STANDBY~~ MODE.

J. ~~HOT STANDBY CONDITION~~ - The reactor is in the ~~HOT STANDBY CONDITION~~ when reactor power is less than or equal to 1 percent of rated. The reactor is in the ~~STARTUP/HOT STANDBY~~ MODE, and the reactor is not in the ~~STARTUP~~ CONDITION. The reactor coolant temperature may be greater than 212° F.

A31

Note that a ~~HOT STANDBY CONDITION~~ cannot exist simultaneously with a ~~STARTUP~~ CONDITION due to the difference in intent. A ~~HOT STANDBY~~ CONDITION exists when the reactor mode switch is placed in the ~~STARTUP/HOT STANDBY~~ position (for example, to comply with an LCO) and power level has been reduced to 1 percent or lower. Anytime control rods are being withdrawn for the purpose of increasing reactor power level, the reactor mode switch has been placed in the ~~STARTUP/HOT STANDBY~~ position, and reactor power level is at or below one percent, a ~~STARTUP~~ CONDITION exists.

A33

K. ~~SHUTDOWN CONDITION~~ - The reactor is in the ~~SHUTDOWN CONDITION~~ when the reactor is in the ~~Shutdown~~ or ~~Refuel~~ Mode.

m2

1. ~~HOT SHUTDOWN CONDITION~~ - The reactor is in the ~~HOT SHUTDOWN~~ CONDITION when reactor coolant temperature is greater than 212° F and the reactor is in the ~~SHUTDOWN~~ CONDITION.

A6

2. ~~COLD SHUTDOWN CONDITION~~ - The reactor is in the ~~COLD SHUTDOWN~~ CONDITION when reactor coolant temperature is equal to or less than 212° F and the reactor is in the ~~SHUTDOWN~~ CONDITION.

A32

L. ~~COLD CONDITION~~ - The reactor is in the ~~COLD~~ CONDITION when reactor coolant temperature is equal to or less than 212° F in any Mode of Operation (except as defined in K.2 above).



M. Mode of Operation - The reactor mode switch position determines the Mode of Operation of the reactor when there is fuel in the reactor vessel, except that the Mode of Operation may remain unchanged when the reactor mode switch is temporarily moved to another position as permitted by the notes. When there is no fuel in the reactor vessel, the reactor is considered not to be in any Mode of Operation or operational condition. The reactor mode switch may then be in any position or may be inoperable.

A34

1. Startup/Hot Standby Mode - The reactor is in the STARTUP/HOT STANDBY MODE when the reactor mode switch is in the "STARTUP/HOT STANDBY" position. This is often referred to as just the STARTUP MODE.
2. Run Mode - The reactor is in the Run Mode when the reactor mode switch is in the "Run" position.
3. Shutdown Mode - The reactor is in the Shutdown Mode when the reactor mode switch is in the "Shutdown" position. (1)

M2

or Refuel

A6

4. Refuel Mode - The reactor is in the Refuel Mode when the reactor mode switch is in the "Refuel" position. (1)

See Justification for Changes for BFN 1ST3 Section 3.10

(1) The reactor mode switch may be placed in any position to perform required tests or maintenance authorized by the shift operations supervisor, provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

(2) The reactor mode switch may be placed in the "Refuel" position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.10.A.5 provided that reactor coolant temperature is equal to or less than 212° F.

(3) The reactor mode switch may be placed in the "Refuel" position while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is OPERABLE.

(4) The reactor mode switch may be placed in the "Startup/Hot Standby" position and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

See Justification for Changes for BFN 1ST3 3.3.2.1



(A1)

THERMAL

(A7)

R. Rated Power - Rated power refers to operation at a reactor power of 3,293 MWt; this is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power.

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

(A35)

1. All nonautomatic containment isolation valves on lines connected to the reactor coolant systems or containment which are not required to be open during accident conditions are closed, except for valves that are open under administrative control as permitted by Specification 3.7.D.
2. At least one door in each airlock is closed and sealed.
3. All automatic containment isolation valves are OPERABLE or each line which contains an inoperable isolation valve is isolated as required by Specification 3.7.D.2.
4. All blind flanges and manways are closed.

P. Secondary Containment Integrity

(A36)

1. Secondary containment integrity means that the required unit reactor zones and refueling zone are intact and the following conditions are met:
 - a) At least one door in each access opening to the turbine building, control bay and out-of-doors is closed.
 - b) The standby gas treatment system is OPERABLE and can maintain 0.25 inches of water negative pressure in those areas where secondary containment integrity is stated to exist.
 - c) All secondary containment penetrations required to be closed during accident conditions are either:
 1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or
 2. Closed by at least one secondary containment automatic isolation valve deactivated in the isolated position.
2. Reactor zone secondary containment integrity means the unit reactor building is intact and the following conditions are met:
 - a) At least one door between any opening to the turbine building, control bay and out-of-doors is closed.



NOV 18 1988

P. Secondary Containment Integrity (Cont'd)

- 2. b) The standby gas treatment system is OPERABLE and can maintain 0.25 inches water negative pressure on the unit zone.
- c) All the unit reactor building ventilation system penetrations required to be closed during accident conditions are either:
 - 1. Capable of being closed by an OPERABLE reactor building ventilation system automatic isolation system, or
 - 2. Closed by at least one reactor building ventilation system automatic isolation valve deactivated in the isolated position.

A36

If it is desirable for operational considerations, a reactor zone may be isolated from the other reactor zones and the refuel zone by maintaining at least one closed door in each common passageway between zones.* Reactor zone safety-related features are not compromised by openings between adjacent units or refuel zone, unless it is desired to isolate a given zone.

- 3. Refuel zone secondary containment integrity means the refuel zone is intact and the following conditions are met:
 - a) At least one door in each access opening to the out-of-doors is closed.
 - b) The standby gas treatment system is OPERABLE and can maintain 0.25 inches water negative pressure on the refuel zone.
 - c) All refuel zone ventilation system penetrations required to be closed during accident conditions are either:
 - 1. Capable of being closed by an OPERABLE refuel zone ventilation system automatic isolation system, or
 - 2. Closed by at least one refuel zone ventilation system automatic isolation valve deactivated in the isolated position.

If it is desirable for operational considerations, the refuel zone may be isolated from the reactor zones by maintaining all hatches in place between the refuel floor and reactor zones and at least one closed door in each access between the refuel zone and the reactor building.* Refuel zone safety-related features are not compromised by openings between the reactor building unless it is desired to isolate a given zone.

*To effectively control zone isolation, all accesses to the affected zone will be locked or guarded to prevent uncontrolled passage to the unaffected zones.

MAY 20 1993

A1 1.0 DEFINITIONS (Cont'd)

A37 Q. Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

A38 R. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

A8 S. CORE ALTERATION - CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Movement of source range monitors, intermediate range monitors, traversing in-core probes, or special movable detectors (including undervessel replacement) is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe location. *(local power range monitors)*
The following exceptions are not considered to be CORE ALTERATIONS:
(a) control rod movement, provided there are no fuel assemblies in the associated core cell. *(position)* *L3*

A39 T. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

A1 U. Thermal Parameters shall be the smallest *(CPR) that exists in the core*
1. Minimum Original Power Ratio (MOPR) - Minimum Critical Power Ratio (MOPR) is the value of the critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in the fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power. *(that)*
by application of the appropriate correlations *divided by*

A40 2. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

A52 3. Core Maximum Fraction of Limiting Power Density (MFLPD) - The MFLPD is the highest ratio, for all fuel assemblies and all axial locations in the core, of the maximum fuel rod power density (kW/ft) for a given fuel assembly and axial location to the limiting fuel rod power density (kW/ft) at that location. *shall be the largest value of the fraction of limiting power density in the core.*

A17 4. Average Planar Linear Heat Generation Rate (APLHGR) - The Average Planar Heat Generation Rate is applicable to a specific planar height and is equal to the sum of the linear heat generation rates for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle. *APLHGR should be*

The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type 1.0-7

LHGRS



(A1) 1.2 DEFINITIONS (Cont'd)

5. CORE MAXIMUM FRACTION OF CRITICAL POWER (CMFCP) - CORE MAXIMUM FRACTION OF CRITICAL POWER is the maximum value of the ratio of the flow-corrected CPR operating limit found in the CORE OPERATING LIMITS REPORT divided by the actual CPR for all fuel assemblies in the core.

V. Instrumentation

1. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors.

2. Channel - A channel is an arrangement of the sensor(s) and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

3. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.

Channel indication and status to other indications or status derived from

4. CHANNEL INSTRUMENT CHECK - An CHANNEL INSTRUMENT CHECK is a qualitative determination of acceptable OPERABILITY by observation of channel-instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

shall be the

assessment

(A11)

5. Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit to insure all components are OPERABLE per design intent. Where practicable, action will go to completion; i.e., pumps will be started and valves operated.

shall be required

all required logic components, (i.e.)

parameter

(A9)

INSERT 1.0-8A

trip units, solid state logic elements, etc.)

6. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

(A41)

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

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



A9

INSERT 1.0-8a

, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.



(A1) 1.0 ~~DEFINITIONS (Cont'd)~~

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

10. Logic - A logic is an arrangement of relays, contacts, and other components that produces a decision output.

(A41)

(a) Initiating - A logic that receives signals from channels and produces decision outputs to the actuation logic.

(b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

A CHANNEL CALIBRATION

11. Channel Calibration - Shall be the adjustment, as necessary, of the channel output such that it responds with necessary range and accuracy to known values of the parameters which the channel monitors. The channel calibration shall encompass the entire channel including alarm and/or trip functions, and shall include the channel functional test. The channel calibration may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated. Non-calibratable components shall be excluded from this requirement, but will be included in channel functional test and source check.

display

(A9)

the required sensor

means of

within the

A CHANNEL FUNCTIONAL TEST

12. Channel Functional Test - Shall be -
a. Analog/Digital Channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
b. Bistable Channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

(A9)

required

interlock display

or actual

and channel failure trips

(L2)

13. (Deleted)

1. DEFINITIONS (Cont'd)

- W. Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water). (A42)
- X. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. (A43)
- Y. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents. (A44)
- Z. Reportable Event - A reportable event shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50. (A45)
- AA. (Deleted)
- BB. Offsite Dose Calculation Manual (ODCM) - Shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.5 and 6.9.1.8. (A18)
- CC. Purge or purging - The controlled process of discharging air or gas from the primary containment to maintain temperature, pressure, humidity, concentration, or other operating condition in such a manner that replacement air or gas is required to purify the containment. (A46)
- DD. (Deleted)
- EE. (Deleted)
- FF. Venting - The controlled process of discharging air or gas from the primary containment to maintain temperature, pressure, humidity, concentration, or other operating condition in such a manner that replacement air or gas is not provided or required. Vent, used in system names, does not imply a venting process. (A20)

1.00 ~~DEFINITIONS~~ (cont'd) (A1)

- (A21) GG. Site Boundary - Shall be that line beyond which the land is not, owned, leased, or otherwise controlled by TVA.
- (A47) HH. Unrestricted Area - 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 industrial, commercial, institutional, or recreational purposes.
- (A12) II. Dose Equivalent I-131 - The ^{microcuries 1 gram} DOSE EQUIVALENT I-131 shall be the concentration of I-131 (in $\mu\text{Ci/gm}$) ^{that} which, alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factor used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites". ^{AEC, 1962,}
- (A48) JJ. Gaseous Waste Treatment System - The charcoal adsorber vessels installed on the discharge of the steam jet air ejector to provide delay to a unit's offgas activity prior to release.
- (A49) KK. Members of the Public - An individual in a controlled or UNRESTRICTED AREA. However, an individual is not a MEMBER OF THE PUBLIC during any period in which the individual receives an occupational dose (as defined in 10 CFR 20).
- (A22) LL. Surveillance - Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual limiting conditions for operation unless otherwise stated in an individual Surveillance Requirements. Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval. It is not intended that this (extension) provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages.
- Performance of a Surveillance Requirement within the specified time interval shall constitute compliance and OPERABILITY requirements for a limiting condition for operation and associated action statements unless otherwise required by these specifications. Surveillance Requirements do not have to be performed on inoperable equipment.
- 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.

1.0^d ~~DEFINITIONS (Cont of~~ (A1)

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.

(A22)

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

MM. Surveillance Requirements for ASME Section XI Pump and Valve Program - Surveillance Requirements for Inservice Testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

1. 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 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 50.55(g)(6)(i).

2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice 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 testing activities

Required frequencies for performing inservice testing activities

(A23)

Weekly
Monthly
Quarterly or every 3 months
Semiannually or every 6 months
Every 9 months
Yearly or annually

At least once per 7 days
At least once per 31 days
At least once per 92 days
At least once per 184 days
At least once per 276 days
At least once per 366 days

- 3. The provisions of Specification 1.0.LL are applicable to the above required frequencies for performing inservice testing activities.
- 4. Performance of the above inservice testing activities shall be in addition to other specified surveillance requirements.
- 5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any technical specification.
- 6. The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, personnel, and sample expansion included in this generic letter.

1.0 DEFINITIONS (Cont'd)

Specification 1.1 FEB 24 1995

cycle specific parameter limits

AD NN. Core Operating Limits Report (COLR) - The COLR is the unit-specific document that provides ~~the core operating limits~~ for the current ~~operating~~ cycle. These cycle specific core operating limits shall be ~~determined~~ for each ~~operating~~ cycle in accordance with Specification ~~6.9.1.7~~. Plant operation within these limits is addressed in individual specifications.

reload

reload

5.6.5

A50 OO. LIMITING CONTROL ROD PATTERN - A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal limit, i.e. operating on a limiting value for APLHGR, LHGR, or MGPR.

A13 INSERT 1 (3 pages)

A6 INSERT 2 (1 page)

A14 INSERT 3 (21 pages)



A13

INSERT 1¹ (Page 1 of 3)

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated conditions within specified Completion Times.
LEAKAGE	LEAKAGE shall be: <ol style="list-style-type: none">a. <u>Identified LEAKAGE</u><ol style="list-style-type: none">1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;b. <u>Unidentified LEAKAGE</u><p>All Leakage into the drywell that is not identified LEAKAGE;</p>c. <u>Total LEAKAGE</u><p>Sum of the identified and unidentified LEAKAGE;</p>d. <u>Pressure Boundary LEAKAGE</u><p>LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.</p>

¹ The definitions found in this insert will be placed in alphabetical order with the other ISTS definitions.

INSERT 1 (Page 2 of 3)

A13

LINEAR HEAT GENERATION
RATE (LHGR)

The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

MODE

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

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 13.10, Refueling Test Program, of the FSAR;
- b. Authorized under the provisions of 10 CFR 50.59; or
- c. Otherwise approved by the Nuclear Regulatory Commission.

SHUTDOWN MARGIN (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:

- a. The reactor is xenon free;
- b. The moderator temperature is 68°F; and
- c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.



A13

STAGGERED TEST BASIS

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

THERMAL POWER

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

TURBINE BYPASS SYSTEM RESPONSE TIME

The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components:

- a. The time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and
- b. The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve.

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



INSERT 2

Ab

Definitions
1.1

Table 1.1-1 (page 1 of 1)
MODES

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

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

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



A14

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 indentions of the logical connectors.

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

EXAMPLES

The following examples illustrate the use of logical connectors.

(continued)

PAGE 20 OF 43



(A14)

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . <u>AND</u> 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)

A14

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . .	
	<u>OR</u>	
	A.2.1 Verify . . .	
	<u>AND</u>	
	A.2.2.1 Reduce . . .	
	<u>OR</u>	
	A.2.2.2 Perform . . .	
	<u>OR</u>	
	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.



A14

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 Times(s).

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

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

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

(continued)

A14

1.3 Completion Times

DESCRIPTION
(continued)

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

- a. Must exist concurrent with the first inoperability; and
- 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 division, subsystem, component or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

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

(continued)

A14

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.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

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

(continued)



INSERT 3 (Page 7 of 21)

A14

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.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	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.

(continued)



INSERT 3 (Page 8 of 21)

1.3 Completion Times

(A14)

EXAMPLES

EXAMPLE 1.3-2 (continued)

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

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)



INSERT 3 (Page 9 of 21)

(A14)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X subsystem inoperable.	A.1 Restore Function X subsystem to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One Function Y subsystem inoperable.	B.1 Restore Function Y subsystem to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO
C. One Function X subsystem inoperable. <u>AND</u> One Function Y subsystem inoperable.	C.1 Restore Function X subsystem to OPERABLE status. <u>OR</u> C.2 Restore Function Y subsystem to OPERABLE status.	12 hours 12 hours

(continued)

1.3 Completion Times

A14

EXAMPLES

EXAMPLE 1.3-3 (continued)

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

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

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

(continued)



INSERT 3 (Page 11 of 21)

(A14)

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.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

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

(continued)



1.3 Completion Times

A14

EXAMPLE
(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.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

(continued)



INSERT 3 (Page 13 of 21)

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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
	<u>OR</u> A.2 Place channel in trip.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

(continued)



1.3 Completion Times

A14

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)



ATF

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 <u>AND</u> Once per 8 hours thereafter
	<u>AND</u> A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

(continued)

A14

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-7 (continued)

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

(A14)

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 Limiting Condition for Operation (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.

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both. Example 1.4-4 discusses these special situations.

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

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

(continued)

(A14)

1.4 Frequency

DESCRIPTION
(continued)

criteria. SR 3.0.4 restrictions would not apply if both the following conditions are satisfied:

- a. The Surveillance is not required to be performed; and
- b. The Surveillance is not required to be met or, even if required to be met, is not known to be failed.

EXAMPLES

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

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

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

(continued)



A14

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-1 (continued)

otherwise modified (refer to Examples 1.4-3 and 1.4-4), 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.

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after $\geq 25\%$ RTP <u>AND</u> 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 $\geq 25\%$ 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 extension allowed by SR 3.0.2.

(continued)



A14

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-2 (continued)

"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.

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>-----NOTE----- Not required to be performed until 12 hours after \geq 25% RTP. -----</p>	
<p>Perform channel adjustment.</p>	<p>7 days</p>

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 \geq 25% RTP to perform the Surveillance. The Surveillance is still considered to be 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 \geq 25% RTP.

(continued)



A14

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

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.

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

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

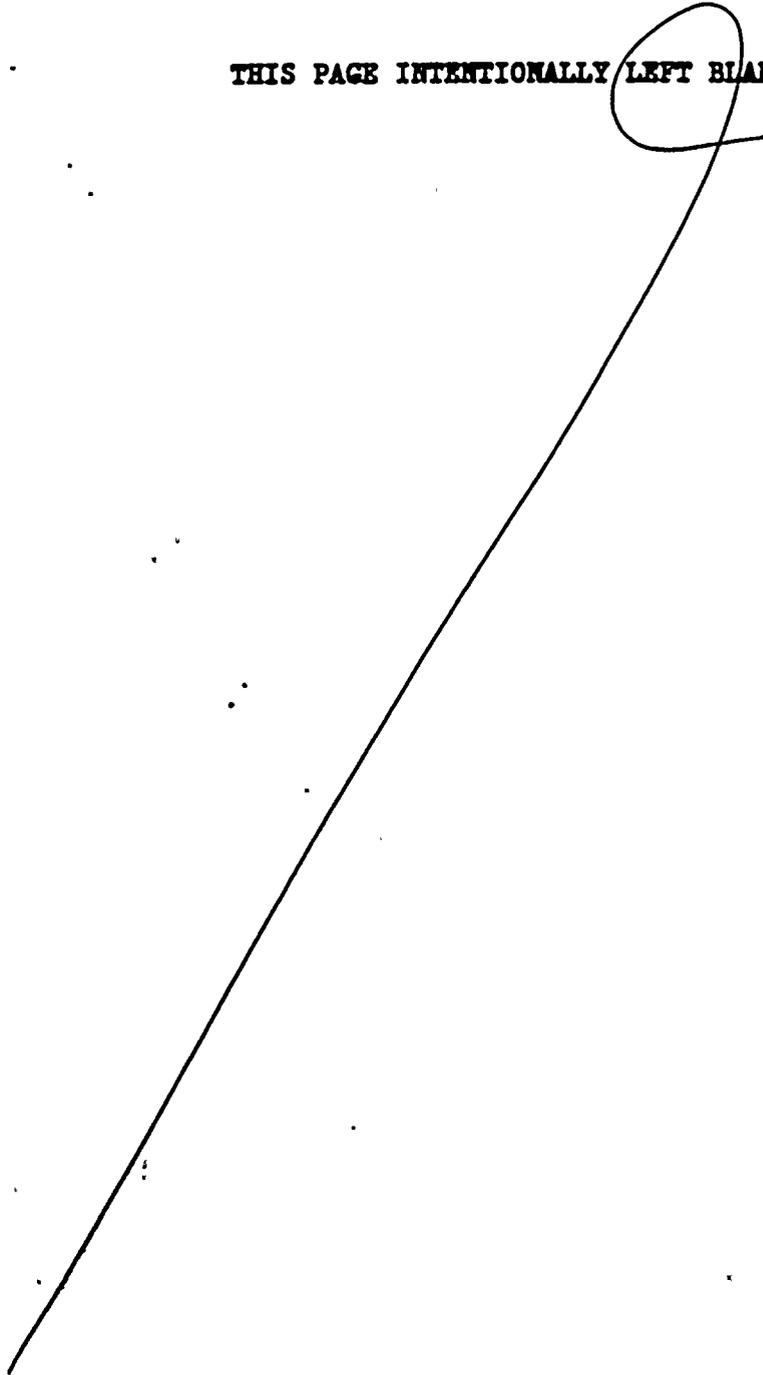
Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to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



Specification 1.1

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Table 1.1

SURVEILLANCE FREQUENCY NOTATION

<u>NOTATION</u>	<u>FREQUENCY</u>
S (Shift)	At least once per 12 hours.
D (Daily)	At least once per normal calendar 24 hour day (midnight to midnight).
W (Weekly)	At least once per 7 days.
M (Monthly)	At least once per 31 days.
Q (Quarterly)	At least once per 3 months or 92 days.
SA (Semi-Annually)	At least once per 6 months or 184 days.
Y (Yearly)	At least once per year or 366 days.
R (Refueling)	At least once per operating cycle.
S/U (Start-Up)	Prior to each reactor startup.
N.A.	Not applicable.
P (Prior)	Completed prior to each release.

(A51)



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UNIT 2
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

A1

1.0 USE AND APPLICATION

1.2

~~DEFINITIONS~~

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

A2

~~The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.~~

A1

Term

~~Definitions~~

1. ~~**Safety Limit** - The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit requires unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.~~
- A25
2. ~~**Limiting Safety System Settings (LSSS)** - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.~~
- A26
3. ~~**Limiting Conditions for Operation (LCO)** - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.~~
- A27
4. ~~In the event a Limiting Condition for Operation and/or associated requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours unless corrective measures are completed that permit operation under the permissible discovery or until the reactor is placed in an operational condition in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications. This provides actions to be taken for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. For example, if a specification calls for two systems (or subsystems) to be operable and provides for explicit requirements if one system (or subsystem) is inoperable, then if both systems (or subsystems) are inoperable the unit is to be in at least Hot Standby in 6 hours and in Cold Shutdown within the following 30 hours if the inoperable condition is not corrected.~~
- A3



NOV 18 1988

A1 1.9

~~DEFINITIONS (Cont'd)~~

2. When a system, subsystem, train, component, or device is determined to be inoperable solely because its onsite power source is inoperable, or solely because its offsite power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided:

A3

(1) its corresponding offsite or diesel 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 these requirements. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least Hot Standby within 6 hours, and in at least Cold Shutdown within the following 30 hours. This definition is not applicable in Cold Shutdown or Refueling. This provision describes what additional conditions must be satisfied to permit operation to continue consistent with the specifications for power sources, when an offsite or onsite power source is not operable. It specifically prohibits operation when one division is inoperable because its offsite or diesel power source is inoperable and a system, subsystem, train, component, or device in another division is inoperable for another reason. This provision permits the requirements associated with individual systems, subsystems, trains, components, or devices to be consistent with the requirements of the associated electrical power source. It allows operation to be governed by the time limit of the requirements associated with the Limiting Condition for Operation for the offsite or diesel power source, not the individual requirements for each system, subsystem, train, component, or device that is determined to be inoperable solely because of the inoperability of its offsite or diesel power source.

A28

~~D. PRIOR TO STARTUP - Prior to withdrawing the first control rod for the purpose of making the reactor critical.~~

~~E. Operable - Operability - A system, subsystem, train, component, or device shall be Operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant~~

A5

~~instrumentation, controls, normal and emergency electrical power resources, cooling or seal water, lubrication, and 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).~~

A29

~~F. Operating - Operating means that a system or component is performing its intended functions in its required manner.~~

A30

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

AMENDMENT NO. 154



1.9 DEFINITIONS (Cont'd)

A6

K. Reactor Power Operation - Reactor power operation is any operation in the STARTUP/HOT STANDBY or RUN MODE with the reactor critical and above 1 percent rated power.

A6

J. STARTUP CONDITION - The reactor is in the STARTUP CONDITION when the withdrawal of control rods for the purpose of making the reactor critical has begun, reactor power is less than or equal to 1 percent of rated, and the reactor is in the STARTUP/HOT STANDBY MODE.

A31

I. HOT STANDBY CONDITION - The reactor is in the HOT STANDBY CONDITION when reactor power is less than or equal to 1 percent of rated. The reactor is in the STARTUP/HOT STANDBY MODE, and the reactor is not in the STARTUP CONDITION. The reactor coolant temperature may be greater than 212° F.

Note that a HOT STANDBY CONDITION cannot exist simultaneously with a STARTUP CONDITION due to the difference in intent. A HOT STANDBY CONDITION exists when the reactor mode switch is placed in the STARTUP/HOT STANDBY position (for example, to comply with an LCO) and power level has been reduced to 1 percent or lower. Anytime control rods are being withdrawn for the purpose of increasing reactor power level, the reactor mode switch has been placed in the STARTUP/HOT STANDBY position, and reactor power level is at or below one percent, a STARTUP CONDITION exists.

A33

H. SHUTDOWN CONDITION - The reactor is in the SHUTDOWN CONDITION when the reactor is in the Shutdown or Refuel Mode.

A6

1. - HOT SHUTDOWN CONDITION - The reactor is in the HOT SHUTDOWN CONDITION when reactor coolant temperature is greater than 212° F and the reactor is in the SHUTDOWN CONDITION.

2. - COLD SHUTDOWN CONDITION - The reactor is in the COLD SHUTDOWN CONDITION when reactor coolant temperature is equal to or less than 212° F and the reactor is in the SHUTDOWN CONDITION.

A32

G. COLD CONDITION - The reactor is in the COLD CONDITION when reactor coolant temperature is equal to or less than 212° F in any Mode of Operation (except as defined in K.2 above).

(A1)

1.0

~~DEFINITIONS (Cont'd)~~

Specification 1.1

APR 30 1993

(A34) M. Mode of Operation - The reactor mode switch position determines the Mode of Operation of the reactor when there is fuel in the reactor vessel, except that the Mode of Operation may remain unchanged when the reactor mode switch is temporarily moved to another position as permitted by the notes. When there is no fuel in the reactor vessel, the reactor is considered not to be in any Mode of Operation or operational condition. The reactor mode switch may then be in any position or may be inoperable.

- (A6)
1. Startup/Hot Standby Mode - The reactor is in the STARTUP/HOT STANDBY MODE when the reactor mode switch is in the "STARTUP/HOT STANDBY" position. This is often referred to as just the STARTUP MODE.
 2. Run Mode - The reactor is in the Run Mode when the reactor mode switch is in the "Run" position.
 3. Shutdown Mode - The reactor is in the Shutdown Mode when the reactor mode switch is in the "Shutdown" position. (1)
(2)(3)(4)
 4. Refuel Mode - The reactor is in the Refuel Mode when the reactor mode switch is in the "Refuel" position. (1)

See Justification for Changes
to BFN 1STS Section 3.10

(1) The reactor mode switch may be placed in any position to perform required tests or maintenance authorized by the shift operations supervisor, provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff. :

(2) The reactor mode switch may be placed in the "Refuel" position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.10.A.5 provided that reactor coolant temperature is equal to or less than 212° F.

(3) The reactor mode switch may be placed in the "Refuel" position while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is OPERABLE.

(4) The reactor mode switch may be placed in the "Startup/Hot Standby" position and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

See Justification for Changes
to BFN 1STS 3.3.2.1



MAY 20 1993

(A1)

1. ~~DEFINITIONS~~ (Governing) (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3293 MWt.

THERMAL

(A7)

~~Rated Power - Rated power refers to operation at a reactor power of 3,293 MWt; this is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power.~~

~~0. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:~~

(A35)

- ~~1. All nonautomatic containment isolation valves on lines connected to the reactor coolant systems or containment which are not required to be open during accident conditions are closed, except for valves that are open under administrative control as permitted by Specification 3.7.D.~~
- ~~2. At least one door in each airlock is closed and sealed.~~
- ~~3. All automatic containment isolation valves are OPERABLE or each line which contains an inoperable isolation valve is isolated as required by Specification 3.7.D.2.~~
- ~~4. All blind flanges and manways are closed.~~

~~P. Secondary Containment Integrity~~

(A36)

- ~~1. Secondary containment integrity means that the required unit reactor zones and refueling zone are intact and the following conditions are met:

 - ~~a) At least one door in each access opening to the turbine building, control bay and out-of-doors is closed.~~
 - ~~b) The standby gas treatment system is OPERABLE and can maintain 0.25 inches of water negative pressure in those areas where secondary containment integrity is stated to exist.~~
 - ~~c) All secondary containment penetrations required to be closed during accident conditions are either:

 - ~~1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or~~
 - ~~2. Closed by at least one secondary containment automatic isolation valve deactivated in the isolated position.~~~~~~
- ~~2. Reactor zone secondary containment integrity means the unit reactor building is intact and the following conditions are met:

 - ~~a) At least one door between any opening to the turbine building, control bay and out-of-doors is closed.~~~~

NOV 18 1988

i.0 (A) DEFINITIONS (Cont'd)

P. Secondary Containment Integrity (Cont'd)

2. b) The standby gas treatment system is OPERABLE and can maintain 0.25 inches water negative pressure on the unit zone.
- c) All the unit reactor building ventilation system penetrations required to be closed during accident conditions are either:
1. Capable of being closed by an OPERABLE reactor building ventilation system automatic isolation system, or
 2. Closed by at least one reactor building ventilation system automatic isolation valve deactivated in the isolated position.

A36

If it is desirable for operational considerations, a reactor zone may be isolated from the other reactor zones and the refuel zone by maintaining at least one closed door in each common passageway between zones.* Reactor zone safety-related features are not compromised by openings between adjacent units or refuel zone, unless it is desired to isolate a given zone.

3. Refuel zone secondary containment integrity means the refuel zone is intact and the following conditions are met:
- a) At least one door in each access opening to the out-of-doors is closed.
 - b) The Standby Gas Treatment System is OPERABLE and can maintain 0.25 inches water negative pressure on the refuel zone.
 - c) All refuel zone ventilation system penetrations required to be closed during accident conditions are either:
 1. Capable of being closed by an OPERABLE refuel zone ventilation system automatic isolation system, or
 2. Closed by at least one refuel zone ventilation system automatic isolation valve deactivated in the isolated position.

If it is desirable for operational considerations, the refuel zone may be isolated from the reactor zones by maintaining all hatches in place between the refuel floor and reactor zones and at least one closed door in each access between the refuel zone and the reactor building.* Refuel zone safety-related features are not compromised by openings between the reactor building unless it is desired to isolate a given zone.

*To effectively control zone isolation, all accesses to the affected zone will be locked or guarded to prevent uncontrolled passage to the unaffected zones.



MAY 20 1993

A1 1.0 DEFINITIONS (cont'd)

A37 Q. Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

A38 R. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

AB The following exceptions are not considered to be CORE ALTERATIONS:
Pa. S. CORE ALTERATION - CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Movement of source range monitors, intermediate range monitors, traversing in-core probes, or special movable detectors (including undervessel replacement); is not and is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe location. (Local power range monitors)
b. Control rod movement, provided that there are 10 fuel assemblies in the associated core cell. (L3)

A39 T. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

A1 U. Thermal Parameters shall be the smallest (CPR) that exists in the core

A15 The 1. Minimum Critical Power Ratio (MCPR) - The Minimum Critical Power Ratio (MCPR) is the value of the critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power. (The) (CPR) that exists in the core
by application of the appropriate correlation(s) divided by

A40 2. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

A52 3. Core Maximum Fraction of Limiting Power Density (MFLPD) - The highest ratio, for all fuel assemblies and all axial locations in the core, of the maximum fuel rod power density (kW/ft) for a given fuel assembly and axial location to the limiting fuel rod power density (kW/ft) at that location.

A17 4. Average Planar Linear Heat Generation Rate (APLHGR) - The Average Planar Heat Generation Rate is applicable to a specific planar height and is equal to the sum of the linear heat generation rates for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle. (APLHGR shall be) (LHGRS)

The MFLPD shall be the largest value of the fraction of limiting power density in the core. The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type. (A52)



(A1)

1.0

DEFINITIONS (Cont'd)

Specification 1.1

OCT 21 1993

5. Core Maximum Fraction of Critical Power (CMFCP) - Core Maximum Fraction of Critical Power is the maximum value of the ratio of the flow-corrected CPR operating limit found in the CORE OPERATING LIMITS REPORT divided by the actual CPR for all fuel assemblies in the core.

(A41)

V. Instrumentation

(A41)

1. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors.
2. Channel - A channel is an arrangement of the sensor(s) and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.
3. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.

channel

channel indication and status to other indications or status derived from

4. Channel Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

shall be the

assessment

(A11)

5. Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit to insure all components are operable per design intent. Where practicable, action will go to completion, i.e., pumps will be started and valves operated.

should be

all required logic components (i.e., all required)

trip units, solid state logic elements, etc.

(A9)

INSECT 1.0-2A

(A41)

6. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

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

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



A9

INSERT 1.0-8a

, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.

OCT 21 1993

A1

1.0-1 DEFINITIONS (Cont'd)

- 9. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.
- 10. Logic - A logic is an arrangement of relays, contacts, and other components that produces a decision output.
 - (a) Initiating - A logic that receives signals from channels and produces decision outputs to the actuation logic.
 - (b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

A41

A CHANNEL CALIBRATION

A9

display

the required sensor

means of

11. Channel Calibration shall be the adjustment, as necessary, of the channel output such that it responds with necessary range and accuracy to known values of the parameters which the channel monitors. The channel calibration shall encompass the entire channel, including alarm and/or trip functions, and shall include the channel functional test. The channel calibration may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated. Non-calibratable components shall be excluded from this requirement, but will be included in channel functional test and source check.

A CHANNEL FUNCTIONAL TEST

- 12. Channel Functional Test shall be:
 - a. Analog/Digital Channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
 - b. Bistable Channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

A9

L2

13. Source Check - Shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source or multiple of sources.

A41



AD 1.0 ~~DEFINITIONS (Cont'd)~~

- W. Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).
A42
- X. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed.
A43
- Y. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents.
A44
- Z. Reportable Event - A reportable event shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.
A45
- AA. (Deleted)
- BB. Offsite Dose Calculation Manual (ODCM) - Shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain:
A48
 - (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs and
 - (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.5 and 6.9.1.8.
- CC. Purge or purging - The controlled process of discharging air or gas from the primary containment to maintain temperature, pressure, humidity, concentration, or other operating condition in such a manner that replacement air or gas is required to purify the containment.
A46
- DD. (Deleted)
- EE. (Deleted)
- FF. Venting - The controlled process of discharging air or gas from the primary containment to maintain temperature, pressure, humidity, concentration, or other operating condition in such a manner that replacement air or gas is not provided or required. Vent, used in system names, does not imply a venting process.
A20



1.00 DEFINITIONS (cont'd) (d1)

- (A21) GG. Site Boundary - Shall be that line beyond which the land is not owned, leased, or otherwise controlled by TVA.
- (A47) HH. Unrestricted Area - 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 industrial, commercial, institutional, or recreational purposes.
- (A12) II. Dose Equivalent I-131 - The ^{microcuries/gram} DOSE EQUIVALENT I-131 shall be the concentration of I-131 (in ^{that} $\mu\text{Ci/gm}$) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites". ^{AEC, 1962,}
- (A48) JJ. Gaseous Waste Treatment System - The charcoal adsorber vessels installed on the discharge of the steam jet air ejector to provide delay to a unit's offgas activity prior to release.
- (A49) KK. Members of the Public - An individual in a controlled or UNRESTRICTED AREA. However, an individual is not a MEMBER OF THE PUBLIC during any period in which the individual receives an occupational dose (as defined in 10 CFR 20).
- (A22) LL. Surveillance - Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual limiting conditions for operation unless otherwise stated in an individual Surveillance Requirements. Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval. It is not intended that this (extension) provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages.
- Performance of a Surveillance Requirement within the specified time interval shall constitute compliance and OPERABILITY requirements for a limiting condition for operation and associated action statements unless otherwise required by these specifications. Surveillance Requirements do not have to be performed on inoperable equipment.
- 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.



1.8 DEFINITIONS (Cont'd) (41)

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.

(A22)

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

MM. Surveillance Requirements for ASME Section XI Pump and Valve Program - Surveillance Requirements for Inservice Testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

(A23)

1. 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 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 50.55(g)(6)(i).
2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these technical specifications:

<u>ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities</u>	<u>Required frequencies for performing inservice testing activities</u>
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

3. The provisions of Specification 1.0.LL are applicable to the above required frequencies for performing inservice testing activities.
4. Performance of the above inservice testing activities shall be in addition to other specified surveillance requirements.
5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any technical specification.
6. The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, personnel, and sample expansion included in this generic letter.



NOV 02 1995

(A1) DEFINITIONS (Cont'd)

XX. Appendix R Safe Shutdown Program

BFN has developed an Appendix R Safe Shutdown Program. This Program is to ensure that the equipment required by the Appendix R Safe Shutdown Analysis is maintained and demonstrated functional as follows:

(A24)

1. The functional requirements of the Safe Shutdown systems and equipment, as well as appropriate compensatory measures should these systems/components be unable to perform their intended function are outlined in Section III of the Program.
2. Testing and monitoring of the Appendix R Safe Shutdown systems and equipment are defined in Section V of the Program.

cycle specific parameter limits

YY. CORE OPERATING LIMITS REPORT (COLR) - The COLR is the unit-specific document that provides ~~the core operating limits~~ for the current operating cycle. These cycle-specific core ~~operating limits~~ shall be determined for each operating cycle in accordance with Specification ~~6.9.1.7~~. Plant operation within these limits is addressed in individual specifications.

(A10) Reload

S.G.S

(A50)

ZZ. Limiting Control Rod Pattern - A limiting control rod pattern shall be a pattern which results in the core being on a thermal limit, i.e. operating on a limiting value for APLHGR, LHGR, or MCDR.

(A13) INSERT 1
(3 pages)

(A6) INSERT 2
(1 page)

(A14) INSERT 3
(21 pages)



(A13)

INSERT 1¹ (Page 1 of 3)

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated conditions within specified Completion Times.
LEAKAGE	LEAKAGE shall be: <ul style="list-style-type: none">a. <u>Identified LEAKAGE</u><ul style="list-style-type: none">1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;b. <u>Unidentified LEAKAGE</u><p>All Leakage into the drywell that is not identified LEAKAGE;</p>c. <u>Total LEAKAGE</u><p>Sum of the identified and unidentified LEAKAGE;</p>d. <u>Pressure Boundary LEAKAGE</u><p>LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.</p>

¹ The definitions found in this insert will be placed in alphabetical order with the other ISTS definitions.



INSERT 1 (Page 2 of 3)

A13

LINEAR HEAT GENERATION
RATE (LHGR)

The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

MODE

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

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 13.10, Refueling Test Program, of the FSAR;
- b. Authorized under the provisions of 10 CFR 50.59; or
- c. Otherwise approved by the Nuclear Regulatory Commission.

SHUTDOWN MARGIN (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:

- a. The reactor is xenon free;
- b. The moderator temperature is 68°F; and
- c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.



INSERT 1 (Page 3 of 3)

A13

STAGGERED TEST BASIS

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

THERMAL POWER

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

TURBINE BYPASS SYSTEM RESPONSE TIME

The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components:

- a. The time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and
- b. The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve.

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



Ab

Table 1.1-1 (page 1 of 1)
MODES

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

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

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

A14

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 indentions of the logical connectors.

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

EXAMPLES

The following examples illustrate the use of logical connectors.

(continued)



(A14)

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . <u>AND</u> 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)

AIY

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . .	
	<u>OR</u>	
	A.2.1 Verify . . .	
	<u>AND</u>	
	A.2.2.1 Reduce . . .	
	<u>OR</u>	
	A.2.2.2 Perform . . .	
<u>OR</u>		
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.

A14

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 Times(s).

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

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

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

(continued)



A14

1.3 Completion Times

DESCRIPTION
(continued)

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

- a. Must exist concurrent with the first inoperability; and
- 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 division, subsystem, component or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

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

(continued)



A14

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.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

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

(continued)

A14

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.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	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.

(continued)



A14

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

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

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)



A14

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X subsystem inoperable.	A.1 Restore Function X subsystem to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One Function Y subsystem inoperable.	B.1 Restore Function Y subsystem to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO
C. One Function X subsystem inoperable. <u>AND</u> One Function Y subsystem inoperable.	C.1 Restore Function X subsystem to OPERABLE status. <u>OR</u> C.2 Restore Function Y subsystem to OPERABLE status.	12 hours 12 hours

(continued)



A14

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

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

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

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

(continued)



(A14)

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.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

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

(continued)



INSERT 3 (Page 12 of 21)

(A14)

1.3 Completion Times

EXAMPLE
(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.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

(continued)

(A14)

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
	<u>OR</u> A.2 Place channel in trip.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

(continued)



A14

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)



A14

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 <u>AND</u> Once per 8 hours thereafter
	<u>AND</u> A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

(continued)



A14

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-7 (continued)

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.

A14

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 Limiting Condition for Operation (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.

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both. Example 1.4-4 discusses these special situations.

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

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

(continued)

A14

1.4 Frequency

DESCRIPTION
(continued)

criteria. SR 3.0.4 restrictions would not apply if both the following conditions are satisfied:

- a. The Surveillance is not required to be performed; and
- b. The Surveillance is not required to be met or, even if required to be met, is not known to be failed.

EXAMPLES

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

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

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

(continued)

(A14)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-1 (continued)

otherwise modified (refer to Examples 1.4-3 and 1.4-4), 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.

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after $\geq 25\%$ RTP <u>AND</u> 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 $\geq 25\%$ 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 extension allowed by SR 3.0.2.

(continued)

(A14)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-2 (continued)

"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.

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>-----NOTE----- Not required to be performed until 12 hours after \geq 25% RTP. -----</p>	
<p>Perform channel adjustment.</p>	<p>7 days</p>

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 \geq 25% RTP to perform the Surveillance. The Surveillance is still considered to be 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 \geq 25% RTP.

(continued)



A14

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

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.

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

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

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



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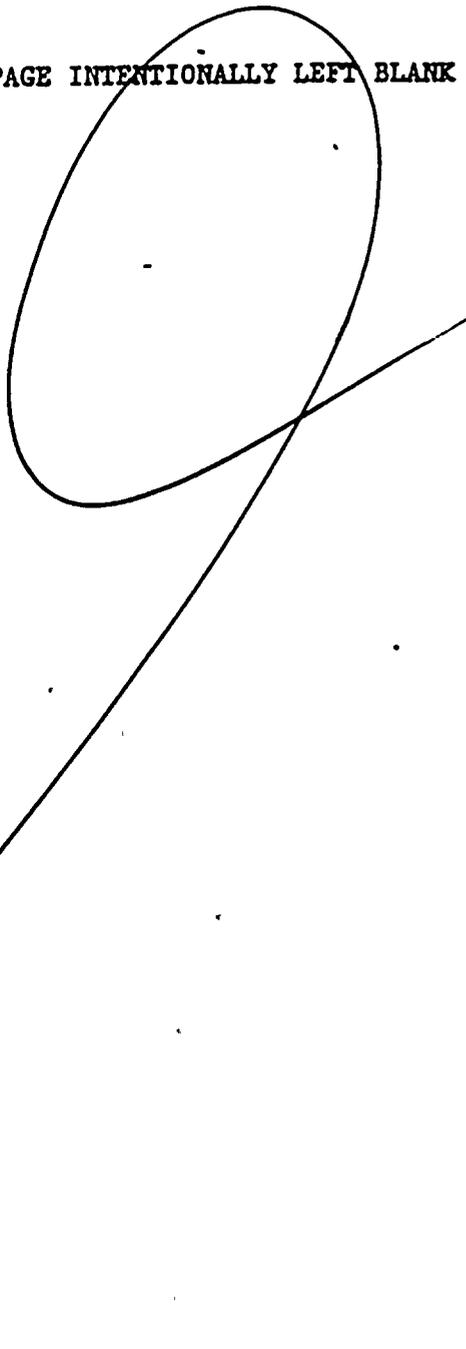




Table 1.1

MAY 19 1989

SURVEILLANCE FREQUENCY NOTATION

NOTATION

FREQUENCY

S	(Shift)	At least once per 12 hours.
D	(Daily)	At least once per normal calendar 24 hour day (midnight to midnight)
W	(Weekly)	At least once per 7 days.
M	(Monthly)	At least once per 31 days.
Q	(Quarterly)	At least once per 3 months or 92 days.
SA	(Semi-Annually)	At least once per 6 months or 184 days.
Y	(Yearly)	At least once per year or 366 days.
R	(Refueling)	At least once per operating cycle.
S/U	(Start-Up)	Prior to each reactor startup.
N.A.		Not applicable.
P	(Prior)	Completed prior to each release.

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UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

(A1) 1.0 USE AND APPLICATION
1.0 DEFINITIONS

NOTE

The defined terms of this section appear in capitalization type and are applicable throughout these TS and Bases

Specification 1.1

(A2)

~~The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.~~

Definitions

A1 (A25)

A. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit requires unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

(A25)

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

(A26)

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

(A27)

D. In the event a Limiting Condition and/or associated requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours unless corrective measures are completed that permit operation under the permissible discovery or until the reactor is placed in an operational condition in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications. This provides action to be taken for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. For example, if a specification calls for two systems (or subsystems) to be operable and provides for explicit requirements if one system (or subsystem) is inoperable, then if both systems (or subsystems) are inoperable the unit is to be in at least Hot Standby in 6 hours and in Cold Shutdown within the following 30 hours if the inoperable condition is not corrected.

(A3)



A1

2. When a system, subsystem, train, component, or device is determined to be inoperable solely because its onsite power source is inoperable, or solely because its offsite 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 offsite or diesel 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 these requirements. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least Hot Standby within 6 hours, and in at least Cold Shutdown within the following 30 hours. This definition is not applicable in Cold Shutdown or Refueling. This provision describes what additional conditions must be satisfied to permit operation to continue consistent with the specifications for power sources, when an offsite or onsite power source is not operable. It specifically prohibits operation when one division is inoperable because its offsite or diesel power source is inoperable and a system, subsystem, train, component, or device in another division is inoperable for another reason. This provision permits the requirements associated with individual systems, subsystems, trains, components, or devices to be consistent with the requirements of the associated electrical power source. It allows operation to be governed by the time limit of the requirements associated with the Limiting Condition For Operation for the offsite or diesel power source, not the individual requirements for each system, subsystem, train, component, or device that is determined to be inoperable solely because of the inoperability of its offsite or diesel power source.

A3

A28

D. ~~PRIOR TO STARTUP~~ - Prior to withdrawing the first control rod for the purpose of making the reactor critical.

A5

and

A29

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

A30

F. Operating - Operating means that a system or component is performing its intended functions in its required manner.

G. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.



NOV 18 1988

(A1) 1.0 DEFINITIONS (Cont'd)

H. Reactor Power Operation - Reactor power operation is any operation in the STARTUP/HOT STANDBY or RUN MODE with the reactor critical and above 1 percent rated power.

A6 I. STARTUP CONDITION - The reactor is in the STARTUP CONDITION when the withdrawal of control rods for the purpose of making the reactor critical has begun, reactor power is less than or equal to 1 percent of rated, and the reactor is in the STARTUP/HOT STANDBY MODE.

J. HOT STANDBY CONDITION - The reactor is in the HOT STANDBY CONDITION when reactor power is less than or equal to 1 percent of rated. The reactor is in the STARTUP/HOT STANDBY MODE, and the reactor is not in the STARTUP CONDITION. The reactor coolant temperature may be greater than 212° F.

A31 Note that a HOT STANDBY CONDITION cannot exist simultaneously with a STARTUP CONDITION due to the difference in intent. A HOT STANDBY CONDITION exists when the reactor mode switch is placed in the STARTUP/HOT STANDBY position (for example, to comply with an LCO) and power level has been reduced to 1 percent or lower. Anytime control rods are being withdrawn for the purpose of increasing reactor power level, the reactor mode switch has been placed in the STARTUP/HOT STANDBY position, and reactor power level is at or below one percent, a STARTUP CONDITION exists.

A33 K. SHUTDOWN CONDITION - The reactor is in the SHUTDOWN CONDITION when the reactor is in the Shutdown or Refuel Mode. (M2)

A6 1. HOT SHUTDOWN CONDITION - The reactor is in the HOT SHUTDOWN CONDITION when reactor coolant temperature is greater than 212° F and the reactor is in the SHUTDOWN CONDITION.

2. COLD SHUTDOWN CONDITION - The reactor is in the COLD SHUTDOWN CONDITION when reactor coolant temperature is equal to or less than 212° F and the reactor is in the SHUTDOWN CONDITION.

A32 L. COLD CONDITION - The reactor is in the COLD CONDITION when reactor coolant temperature is equal to or less than 212° F in any Mode of Operation (except as defined in K.2 above).

APR 30 1993

A1
1.0~~DEFINITIONS (Cont'd)~~

M. Mode of Operation - The reactor mode switch position determines the Mode of Operation of the reactor when there is fuel in the reactor vessel, except that the Mode of Operation may remain unchanged when the reactor mode switch is temporarily moved to another position as permitted by the notes. When there is no fuel in the reactor vessel, the reactor is considered not to be in any Mode of Operation or operational condition. The reactor mode switch may then be in any position or may be inoperable.

A34

1. Startup/Hot Standby Mode - The reactor is in the STARTUP/HOT STANDBY MODE when the reactor mode switch is in the "STARTUP/HOT STANDBY" position. This is often referred to as just the STARTUP MODE.

A6

2. Run Mode - The reactor is in the Run Mode when the reactor mode switch is in the "Run" position.

3. Shutdown Mode - The reactor is in the Shutdown Mode when the reactor mode switch is in the "Shutdown" position. (1)
(2)(3)(4)

A6

4. Refuel Mode - The reactor is in the Refuel Mode when the reactor mode switch is in the "Refuel" position. (1)

See Justification for Changes for BFN ISTS Section 3.10

(1) The reactor mode switch may be placed in any position to perform required tests or maintenance authorized by the shift operations supervisor, provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

(2) The reactor mode switch may be placed in the "Refuel" position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.10.A.5 provided that reactor coolant temperature is equal to or less than 212° F.

(3) The reactor mode switch may be placed in the "Refuel" position while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is OPERABLE.

(4) The reactor mode switch may be placed in the "Startup/Hot Standby" position and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

See Justification for Changes for BFN ISTS 3.3.2.1



1.0 DEFINITIONS (Cont'd)
THERMAL (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3293 MWt

Specification 1.1

MAY 20 1993

N. Rated Power - Rated power refers to operation at a reactor power of 3,293 MWt; this is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power.

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

- 1. All nonautomatic containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed, except for valves that are open under administrative control as permitted by Specification 3.7.D.
- 2. At least one door in each airlock is closed and sealed.
- 3. All automatic containment isolation valves are OPERABLE or each line which contains an inoperable isolation valve is isolated as required by Specification 3.7.D.2.
- 4. All blind flanges and manways are closed.

P. Secondary Containment Integrity

- 1. Secondary containment integrity means that the required unit reactor zones and refueling zone are intact and the following conditions are met:
 - a) At least one door in each access opening to the turbine building, control bay and out-of-doors is closed.
 - b) The standby gas treatment system is OPERABLE and can maintain 0.25 inches of water negative pressure in those areas where secondary containment integrity is stated to exist.
 - c) All secondary containment penetrations required to be closed during accident conditions are either:
 - 1. Capable of being closed by an OPERABLE secondary containment automatic isolation position, or
 - 2. Closed by at least one secondary containment automatic isolation valve deactivated in the isolated position.
- 2. Reactor zone secondary containment integrity means the unit reactor building is intact and the following conditions are met:
 - a) At least one door between any opening to the turbine building, control bay and out-of-doors is closed.



NOV 18 1988

P. Secondary Containment Integrity (Cont'd)

2. b) The Standby Gas Treatment System is OPERABLE and can maintain 0.25 inches water negative pressure on the unit zone.
- c) All the unit reactor building ventilation system penetrations required to be closed during accident conditions are either:
 1. Capable of being closed by an OPERABLE reactor building ventilation system automatic isolation system, or
 2. Closed by at least one reactor building ventilation system automatic isolation valve deactivated in the isolated position.

A36

If it is desirable for operational considerations, a reactor zone may be isolated from the other reactor zones and the refuel zone by maintaining at least one closed door in each common passageway between zones.* Reactor zone safety-related features are not compromised by openings between adjacent units or refuel zone, unless it is desired to isolate a given zone.

3. Refuel zone secondary containment integrity means the refuel zone is intact and the following conditions are met:
 - a) At least one door in each access opening to the out-of-doors is closed.
 - b) The standby gas treatment system is OPERABLE and can maintain 0.25 inches water negative pressure on the refuel zone.
 - c) All refuel zone ventilation system penetrations required to be closed during accident conditions are either:
 1. Capable of being closed by an OPERABLE refuel zone ventilation system automatic isolation system, or
 2. Closed by at least one refuel zone ventilation system automatic isolation valve deactivated in the isolated position.

If it is desirable for operational considerations, the refuel zone may be isolated from the reactor zones by maintaining all hatches in place between the refuel floor and reactor zones and at least one closed door in each access between the refuel zone and the reactor building.* Refuel zone safety-related features are not compromised by openings between the reactor building unless it is desired to isolate a given zone.

*To effectively control zone isolation, all accesses to the affected zone will be locked or guarded to prevent uncontrolled passage to the unaffected zones.

PAGE 7 OF 43



^(A37) Q. Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

^(A38) R. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

^(A8) S. CORE ALTERATION - CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Movement of source range monitors, intermediate range monitors, traversing in-core probes, or special movable detectors (including undervessel replacement); is not and considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe location. ^(local Power Range monitors) ^(position) ^(b. control rod movement, provided there are no fuel assemblies in the associated core cell) ^(L3)

The following exceptions are not considered to be CORE ALTERATIONS:
P a.

^(A39) T. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

^(A1) U. Thermal Parameters shall be the smallest ^{(CPR) that exist in the core}

^(A15) 1. Minimum Critical Power Ratio (MCPR) - ^{The} Minimum Critical Power Ratio (MCPR) is the value of the critical power ratio associated with the most limiting assembly in the reactor core. ^{The} Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly which is calculated to cause some point in the assembly to ^(that) experience boiling transition, to the actual assembly operating power. ^(divided by) ^{(by application of the appropriate correction(s))}

^(A40) 2. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

^(A52) 3. Core Maximum Fraction of Limiting Power Density (CMFLPD) - The highest ratio, for all fuel assemblies and all axial locations in the core, of the maximum fuel rod power density (kW/ft) for a given fuel assembly and axial location to the limiting fuel rod power density (kW/ft) at that location.

^(A17) 4. Average Planar Linear Heat Generation Rate (APLHGR) - ^(APLHGR shall be) The Average Planar Heat Generation Rate is applicable to a specific planar height and is equal to the sum of the linear heat generation rates for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle. ^(LHGRs)

The MFLPD shall be the largest value of the fraction of limiting power density in the core. The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

FEB 24 1995

(AI) 1.0 DEFINITIONS (Cont'd)

5. CORE MAXIMUM FRACTION OF CRITICAL POWER (CMFCP) - CORE MAXIMUM FRACTION OF CRITICAL POWER is the maximum value of the ratio of the flow-corrected CPR operating limit found in the CORE OPERATING LIMITS REPORT divided by the actual CPR for all fuel assemblies in the core. (A41)

V. Instrumentation

1. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors. (A41)

2. Channel - A channel is an arrangement of the sensor(s) and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

3. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.

4. CHANNEL Instrument Check - An CHANNEL instrument check is qualitative ^{shall be the} ^{assessment} determination of acceptable ~~OPERABILITY~~ ^{assessment} by observation of ~~instrument~~ ^{channel} behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same ~~variable~~ ^{parameter}. (A1)

channel indication and status to other indications or status derived from

5. Logic System Functional Test - A logic system functional test means a test of ^{all required logic components (i.e., all required)} ~~all relays and contacts~~ of a logic circuit to ~~insure all components are OPERABLE per design intent~~ ^{shall be}. Where practicable, action will go to completion; i.e., pumps will be started and valves operated. ^{INSECT 1.0-8A} ^{trip units, solid state logic elements, etc.} (A9)

6. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems. (A41)

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

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



INSERT 1.0-8a

(A9)

, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.



FEB 24 1995

(A1) 1.0 ~~Definitions~~ (Cont'd)

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

10. Logic - A logic is an arrangement of relays, contacts, and other components that produces a decision output.

A412

(a) Initiating - A logic that receives signals from channels and produces decision outputs to the actuation logic.

(b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

A CHANNEL CALIBRATION

11. Channel Calibration - Shall be the adjustment, as necessary, of the channel output such that it responds with necessary range and accuracy to known values of the parameters which the channel monitors. The channel calibration shall encompass the entire channel, including alarm and/or trip functions, and shall include the channel functional test. The channel calibration may be performed by any series of sequential, overlapping or total channel tests such that the entire channel is calibrated. Non-calibratable components shall be excluded from this requirement, but will be included in channel functional test and source check.

A9

display
the required sensor
means of

A CHANNEL FUNCTIONAL TEST

12. Channel Functional Test - Shall be:

A9

(1) Analog/Digital Channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel tests so that the entire channel is tested.

Required Interlock Display

L2

(2) Bistable Channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

13. (Deleted)



1. ~~1.0~~ ~~1.0-10~~ (cont'd) (A1)

W. Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or components to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water). (A42)

X. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. (A43)

Y. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents. (A44)

Z. Reportable Event - A reportable event shall be any of those conditions specified in section 50.73 to 10 CFR Part 50. (A45)

AA. (Deleted)

BB. Offsite Dose Calculation Manual (ODCM) - Shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.5 and 6.9.1.8. (A18)

CC. Purge or purging - The controlled process of discharging air or gas from the primary containment to maintain temperature, pressure, humidity, concentration, or other operating condition in such a manner that replacement air or gas is required to purify the containment. (A46)

DD. (Deleted)

EE. (Deleted)

FF. Venting - The controlled process of discharging air or gas from the primary containment to maintain temperature, pressure, humidity, concentration, or other operating condition in such a manner that replacement air or gas is not provided or required. Vent, used in system names, does not imply a venting process. (A20)

1.0 ~~DESCRIPTION (cont'd)~~ (A1)

- (A21) GG. Site Boundary - Shall be that line beyond which the land is not owned, leased, or otherwise controlled by TVA.
- (A47) HH. Unrestricted Area - 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 industrial, commercial, institutional, or recreational purposes.
- (A12) II. Dose Equivalent I-131 - The ^{microcuries/gram} DOSE EQUIVALENT I-131 ^{that} shall be the concentration of I-131 (in $\mu\text{Ci/gm}$) which, alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factor used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites", ^{AEC, 1962,}
- (A48) JJ. Gaseous Waste Treatment System - The charcoal adsorber vessels installed on the discharge of the steam jet air ejector to provide delay to a unit's offgas activity prior to release.
- (A46) KK. Members of the Public - An individual in a controlled or UNRESTRICTED AREA. However, an individual is not a MEMBER OF THE PUBLIC during any period in which the individual receives an occupational dose (as defined in 10 CFR 20).
- (A22) LL. Surveillance - Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual limiting conditions for operation unless otherwise stated in an individual Surveillance Requirements. Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval. It is not intended that this (extension) provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages.

Performance of a Surveillance Requirement within the specified time interval shall constitute compliance and OPERABILITY requirements for a limiting condition for operation and associated action statements unless otherwise required by these specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

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.



1.0 DEFINITIONS (Cont'd) (A1)

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.

(A22)

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

MM. Surveillance Requirements for ASME Section XI Pump and Valve Program - Surveillance Requirements for Inservice Testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

(A23)

1. 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).
2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice 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 testing activities	Required frequencies for performing inservice 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

3. The provisions of Specification 1.0.LL are applicable to the above required frequencies for performing inservice testing activities.
4. Performance of the above inservice testing activities shall be in addition to other specified surveillance requirements.
5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any technical specification.
6. The inservice inspection program for piping identified in ASME Generic Letter 89-01 shall be performed in accordance with the staff positions on schedule, methods, personnel, and sample expansion included in this generic letter.



NOV 02 1995

1.0 ~~DEFINITIONS (Cont'd)~~ e

(A1)

NN. Appendix R Safe Shutdown Program - BFN has developed an Appendix R Safe Shutdown Program. This program is to ensure that the equipment required by Appendix R Safe Shutdown Analysis is maintained and demonstrated functional as follows:

(A24)

1. The functional requirements of the Safe Shutdown Systems and equipment, as well as appropriate compensatory measures should these systems/components be unable to perform their intended function are outlined in Section III of the program.
2. Testing and monitoring of the Appendix R Safe Shutdown systems and equipment are defined in Section V of the Program.

(A10)

00. CORE OPERATING LIMITS REPORT (COLR) - The COLR is the unit-specific document that provides the core operating limits for the current operating cycle. These cycle specific core operating limits shall be determined for each operating cycle in accordance with Specification 6.9.1.7. Plant operation within these limits is addressed in individual specifications.

(reload)

(S.G.S)

(reload)

(A50)

PP. LIMITING CONTROL ROD PATTERN - A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal limit, i.e. operating on a limiting value for APLGR, LHGR, or MCPR.

(A13)

INSERT 1
(3 PAGES)

(A6)

INSERT 2
(1 PAGE)

(A14)

INSERT 3
21 Pages



A13

INSERT 1¹ (Page 1 of 3)

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated conditions within specified Completion Times.
LEAKAGE	LEAKAGE shall be: <ol style="list-style-type: none">a. <u>Identified LEAKAGE</u><ol style="list-style-type: none">1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;b. <u>Unidentified LEAKAGE</u><p>All Leakage into the drywell that is not identified LEAKAGE;</p>c. <u>Total LEAKAGE</u><p>Sum of the identified and unidentified LEAKAGE;</p>d. <u>Pressure Boundary LEAKAGE</u><p>LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.</p>

1

The definitions found in this insert will be placed in alphabetical order with the other ISTS definitions.



INSERT 1 (Page 2 of 3)

A13

LINEAR HEAT GENERATION RATE (LHGR)

The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

MODE

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

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 13.10, Refueling Test Program, of the FSAR;
- b. Authorized under the provisions of 10 CFR 50.59; or
- c. Otherwise approved by the Nuclear Regulatory Commission.

SHUTDOWN MARGIN (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:

- a. The reactor is xenon free;
- b. The moderator temperature is 68°F; and
- c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.



A13

STAGGERED TEST BASIS

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

THERMAL POWER

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

TURBINE BYPASS SYSTEM RESPONSE TIME

The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components:

- a. The time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and
- b. The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve.

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



Ab

Table 1.1-1 (page 1 of 1)
MODES

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

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

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



A14

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 indentions of the logical connectors.

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

EXAMPLES The following examples illustrate the use of logical connectors.

(continued)



(A14)

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . <u>AND</u> 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)

(A14)

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . .	
	<u>OR</u>	
	A.2.1 Verify . . .	
	<u>AND</u>	
	A.2.2.1 Reduce . . .	
	<u>OR</u>	
	A.2.2.2 Perform . . .	
	<u>OR</u>	
	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.



AIY

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 Times(s).

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

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

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

(continued)

A14

1.3 Completion Times

DESCRIPTION
(continued)

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

- a. Must exist concurrent with the first inoperability; and
- 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 division, subsystem, component or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

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

(continued)



A14

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.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

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

(continued)



AIY

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.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	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.

(continued)

1.3 Completion Times

A14

EXAMPLES

EXAMPLE 1.3-2 (continued)

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

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)

A14

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X subsystem inoperable.	A.1 Restore Function X subsystem to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One Function Y subsystem inoperable.	B.1 Restore Function Y subsystem to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO
C. One Function X subsystem inoperable. <u>AND</u> One Function Y subsystem inoperable.	C.1 Restore Function X subsystem to OPERABLE status. <u>OR</u> C.2 Restore Function Y subsystem to OPERABLE status.	12 hours 12 hours

(continued)



A14

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

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

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

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

(continued)



A14

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.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

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

(continued)

A14

1.3 Completion Times

EXAMPLE
(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.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

(continued)

(A14)

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
	<u>OR</u> A.2 Place channel in trip.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

(continued)



1.3 Completion Times

A14

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)

(A14)

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 <u>AND</u> Once per 8 hours thereafter
	<u>AND</u> A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

(continued)



1.3 Completion Times

A14

EXAMPLES

EXAMPLE 1.3-7 (continued)

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

A14

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 Limiting Condition for Operation (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.

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both. Example 1.4-4 discusses these special situations.

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

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

(continued)



A14

1.4 Frequency

DESCRIPTION (continued) criteria. SR 3.0.4 restrictions would not apply if both the following conditions are satisfied:

- a. The Surveillance is not required to be performed; and
- b. The Surveillance is not required to be met or, even if required to be met, is not known to be failed.

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

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

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

(continued)



AIY

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-1 (continued)

otherwise modified (refer to Examples 1.4-3 and 1.4-4), 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.

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP <u>AND</u> 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 extension allowed by SR 3.0.2.

(continued)



(A14)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-2 (continued)

"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.

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>-----NOTE----- Not required to be performed until 12 hours after \geq 25% RTP. -----</p>	
<p>Perform channel adjustment.</p>	<p>7 days</p>

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 \geq 25% RTP to perform the Surveillance. The Surveillance is still considered to be 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 \geq 25% RTP.

(continued)



A14

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

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.

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

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

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



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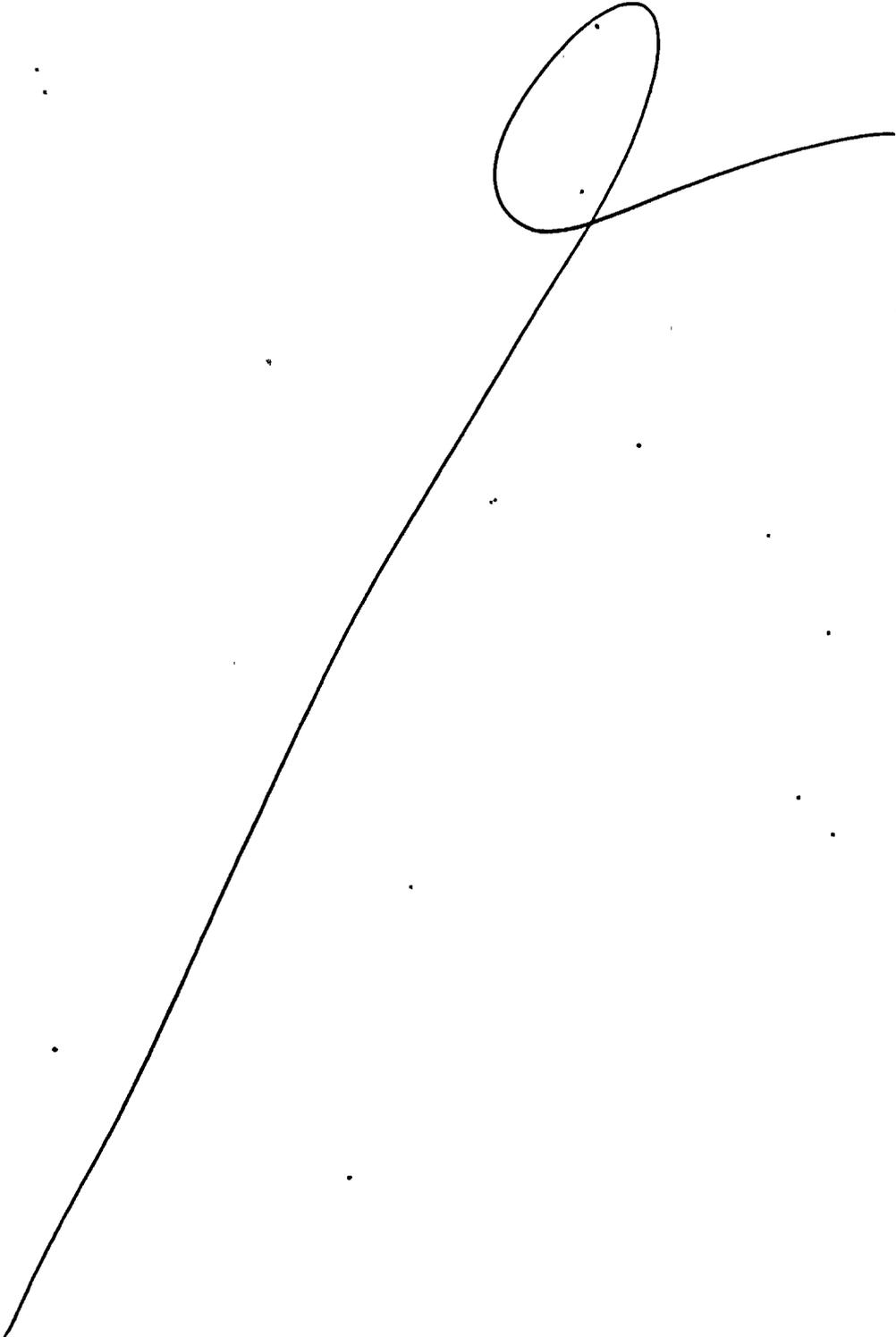




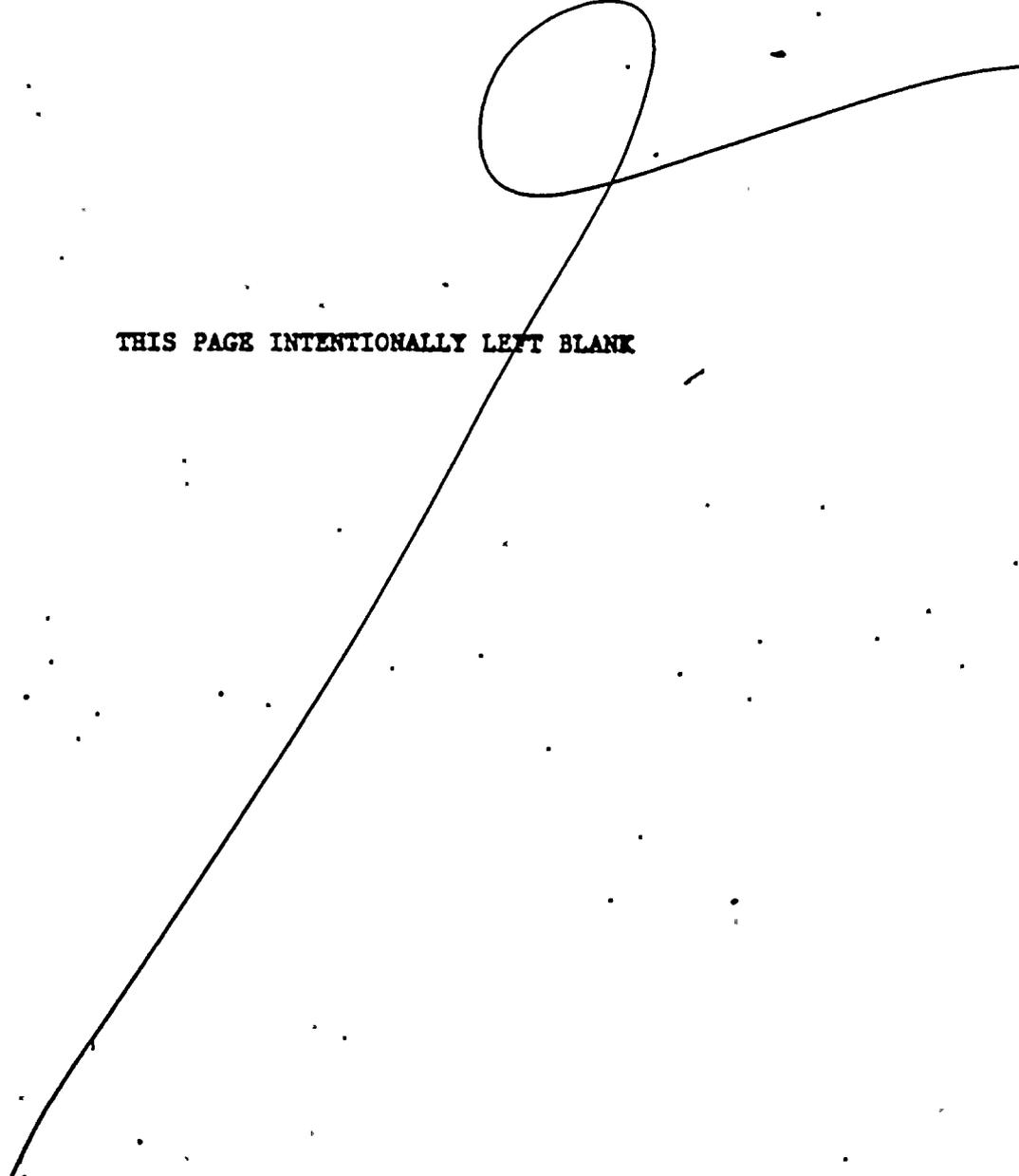
Table 1.1

SURVEILLANCE FREQUENCY NOTATION

<u>NOTATION</u>	<u>FREQUENCY</u>
S (Shift)	At least once per 12 hours.
D (Daily)	At least once per normal calendar 24 hour day (midnight to midnight).
W (Weekly)	At least once per 7 days.
M (Monthly)	At least once per 31 days.
Q (Quarterly)	At least once per 3 months or 92 days.
SA (Semi-Annually)	At least once per 6 months or 184 days.
Y (Yearly)	At least once per year or 366 days.
R (Refueling)	At least once per operating cycle.
S/U (Start-Up)	Prior to each reactor startup.
N.A.	Not applicable.
P (Prior)	Completed prior to each release.

(ASI)





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**JUSTIFICATION FOR CHANGES
SECTION 1.0 - USE AND APPLICATION**

ADMINISTRATIVE CHANGES

- A1 All Reformatting and renumbering is in accordance with the BWR Standard Technical specifications, NUREG-1433. As a result, the Technical Specifications should be more readily readable, and therefore understandable, by plant operators as well as other users. During this reformatting and renumbering process, no technical changes (either actual or interpretational) to the Technical Specifications were made unless they were identified and justified.
- A2 A note was added to Section 1.1, "Definitions," in order to clarify that the defined terms will appear capitalized and are applicable throughout the Technical Specifications and Bases. This addition is administrative in that it clarifies the use of the definitions throughout the Technical Specifications without changing the intent of any Technical Specification. This change maintains the consistency between the BFN ISTS and the BWR/4 ISTS.
- A3 C.1 of Section 1.0 is being reworded and moved to Section 3.0 of the BFN ISTS. This change maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The movement of this portion of the definition is administrative. Any changes to 1.0.C.1 are justified in the change package for Section 3.0.
- A4 Not used.
- A5 The rewording and title capitalization of the "Operable - Operability" definition is administrative because the meaning was not changed. During the BWR/4 ISTS development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. This change maintains the consistency between the BFN ISTS and the BWR/4 ISTS.

A specific change to this definition is changing the "and" to an "or" in "normal and emergency power sources." This is an administrative change because currently the definition along with 1.0.C.2 requires only one source to be operable as long as the redundant systems, subsystems, trains, components, and devices are Operable. Current Specification 1.0.C.2 requirements are incorporated into proposed LCO 3.8.1 ACTIONS for when a diesel or offsite power source is inoperable. Thus, the new requirements are effectively the same as the current requirements. In LCO 3.8.1, new times have been provided to perform the determination of redundant feature Operability. These changes are discussed in the Justification for Changes for LCO 3.8.1.



JUSTIFICATION FOR CHANGES
SECTION 1.0 - USE AND APPLICATION

ADMINISTRATIVE CHANGES (CONTINUED)

- A6 The definitions of "Reactor Power Operation," "Startup Conditions," "Hot Shutdown Conditions," "Cold Shutdown Conditions," "Startup/Hot Standby Mode," "Run Mode," "Shutdown Mode," and "Refuel Mode" are incorporated into a "MODES" table (Table 1.1-1 of the BFN ISTS) with column headings: Mode, Title, Reactor Mode Switch Position, and Average Reactor Coolant Temperature. This change makes the modes more definitive, which decreases the likelihood of being in more than one mode. This change maintains the consistency between the BFN ISTS and the BWR/4 ISTS.
- A7 The title "Rated Power" is changed to "RATED THERMAL POWER." This makes the title more accurately match the definition which discusses the thermal power. The title change and changes to the wording make the definition consistent with the BWR/4 ISTS. The portion of the definition dealing with design power was deleted because it is superfluous to the definition of RATED THERMAL POWER. During the BWR/4 ISTS development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications.
- A8 The rewording and title capitalization of the "Core Alteration" definition is administrative because the meaning was not changed. The provision added that allows control rod movement with no fuel assemblies in the core cell to not be considered a Core Alteration is less restrictive and is discussed in Comment L3 below. During the BWR/4 ISTS development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. This change maintains the consistency between the BFN ISTS and the BWR/4 ISTS.
- A9 The definitions of Channel Calibration and Logic System Functional Test (LSFT) were changed. The Channel Calibration definition was clarified to exclude Non-calibratable components. The Non-calibratable components will be included in the channel functional test and source test. The definition of LSFT was revised to remove the requirement to include the sensor and the end device. The end device will be tested during the system functional testing requirements of the affected LCO (e.g., proposed SR 3.5.1.9, which tests to ensure an ECCS pump starts automatically on an initiation signal). Since any of the tests can be credited for performance in parts, as long as the whole channel is tested, it does not matter when the sensor and end device are tested (i.e., with the Channel Calibration, the LSFT, or the system functional test). Thus, the accumulation of both of these changes results in an administrative change.

PAGE 2 OF 13

**JUSTIFICATION FOR CHANGES
SECTION 1.0 - USE AND APPLICATION**

ADMINISTRATIVE CHANGES (CONTINUED)

In addition, the word "required" has been added to the Channel Calibration, Channel Functional Test, and the Logic System Functional Test definitions. As a requirement for Operability of a Technical Specification channel, not all channels will have a required sensor or alarm function. Conversely, some channels may have required display function. This is the intent of existing wording, and therefore, the revised wording is proposed to more accurately reflect this intent, consistent with the current licensing basis and BWR/4 ISTS, NUREG-1433.

- A10 Editorial changes to make consistent with the BWR/4 STS, NUREG-1433. During ISTS development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the TS.
- A11 The "Instrument Check" definition is reworded and the title was changed to "CHANNEL CHECK." This is an administrative change because the meaning was not changed. During the BWR/4 ISTS English development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. This change maintains the consistency between the BFN ISTS and the BWR/4 ISTS.
- A12 The rewording and title capitalization of the "Dose Equivalent I-131" definition is administrative because the meaning was not changed. During the BWR/4 ISTS development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. This change maintains the consistency between the BFN ISTS and the BWR/4 ISTS.
- A13 Nine definitions are added to the BFN Technical Specifications. These definitions were added for consistency with the BWR/4 ISTS. These definitions are used throughout the BFN ISTS and in the current BFN Technical Specifications. The defined terms are used in the LCOs, Surveillance Requirements, and Bases of the Technical Specifications and were defined for the convenience of the users of the Technical Specifications. The inclusion of these definitions are deemed administrative and have no impact on their own.
- A14 Sections are being added to the Technical Specifications. These additions aid in the understanding and use of the new standard Technical Specifications format and style of presentation. Some conventions in applying the Technical Specifications to unique situations have previously been the subject of debate and interpretation by the licensee and the NRC Staff. Because the guidance in these proposed sections is presented in the BWR/4 Standard Technical Specifications, NUREG-1433, as



**JUSTIFICATION FOR CHANGES
SECTION 1.0 - USE AND APPLICATION**

ADMINISTRATIVE CHANGES (CONTINUED)

approved by the NRC Staff, and the guidance is not a specific deviation from anything in the existing Technical Specifications, these additions are considered to be administrative. The added sections are as follows:

SECTION 1.2 - Logical Connectors

Proposed Section 1.2 provides specific examples of the logical connectors "AND" and "OR" and the numbering sequence associated with their use. This revision is being proposed consistent with the BWR/4 Standard Technical Specification, NUREG-1433.

SECTION 1.3 - Completion Times

Proposed Section 1.3 provides proper use and interpretation of Completion Times. The proposed section also provides specific examples that aid the user in understanding Completion Times. The proposed Completion Times Section is consistent with the BWR/4 Standard Technical specification, NUREG-1433.

SECTION 1.4 - Frequency

Proposed Section 1.4 provides proper use and interpretation of the Surveillance frequency. The proposed section also provides specific examples that aid the user in understanding Surveillance Frequency. The proposed Frequency Section is consistent with the BWR/4 Standard Technical Specification, NUREG-1433.

- A15 The rewording and title capitalization of the "Minimum Critical Power Ratio" definition is administrative because the meaning was not changed. During the BWR/4 ISTS development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. This change maintains the consistency between the BFN ISTS and the BWR/4 ISTS.
- A16 Not used.
- A17 The rewording and title capitalization of the "Average Planar Linear Heat Generation Rate (APLHGR)" definition is administrative because the meaning was not changed. During the BWR/4 ISTS development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. This change maintains the consistency between the BFN ISTS and the BWR/4 ISTS.



**JUSTIFICATION FOR CHANGES
SECTION 1.0 - USE AND APPLICATION**

ADMINISTRATIVE CHANGES (CONTINUED)

- A18 The "Offsite Dose Calculations Manual (ODCM)" definition is moved to Section 5.0 of the BFN ISTS with some wording changes to make it consistent with the BWR/4 ISTS. This is an administrative change because the definition is being moved to another section and has no impact on any other definition nor does it change the intent of any Technical Specification. Any technical changes will be justified in the change package for Section 5.0.
- A19 Not used.
- A20 The definition of "Venting" was deleted because it is not used in the LCOs or Surveillance Requirements of the BFN ISTS. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is administrative with no impact of its own.
- A21 The "Site Boundary" definition is reworded and moved to Section 4.0, "Design Features." In Section 4.0, a map depicts the site boundary. This change maintains the consistency between the BFN ISTS and the BWR/4 ISTS. This is an administrative change because the definition is being moved, with wording changes, to another section and has no impact on any other definition nor does it change the intent of any Technical Specification.
- A22 The requirements specified by the definition of "Surveillance" are moved to the BFN ISTS Section 3.0, "Surveillance Requirement (SR) Applicability." The requirements were reworded and incorporated into SR 3.0.1, SR 3.0.2, and SR 3.0.3. This is an administrative change because the requirements are being moved to another Technical Specifications section and has no impact on any other definition nor does it change the intent of any Technical Specification. Any technical change will be justified in the change package for Section 3.0. This change maintains the consistency between the BFN ISTS and the BWR/4 ISTS.
- A23 The requirements specified by the definition of "Surveillance Requirements for ASME Section XI Pump and Valve Program" are moved to Section 5.0, "Administrative Controls." This is an administrative change because it is being relocated, with wording changes, to another section and has no impact on any other definition nor does it change the intent of any Technical Specification. This change maintains the consistency between the BFN ISTS and the BWR/4 ISTS. Any technical changes will be justified in the change package for Section 5.0.



**JUSTIFICATION FOR CHANGES
SECTION 1.0 - USE AND APPLICATION**

ADMINISTRATIVE CHANGES (CONTINUED)

- A24 The requirements specified by the definition of "Appendix R Safe Shutdown Program" (Unit 2 and 3 only) are contained in the existing BFN Appendix R safe shutdown program. This is an administrative change because this definition duplicates the 10 CFR 50, Appendix R requirements. This change has no impact on other definitions nor does it change the intent of any Technical Specification. This change maintains the consistency between the BFN ISTS and the BWR/4 ISTS.
- A25 The "Safety Limit" definition is deleted because the definition already exists in 10 CFR 50.36 and does not need to be repeated in the Technical Specifications. The use of Safety Limits in the proposed BFN ISTS Section 2.0, Safety Limits, clearly depicts they are the limits below which the reasonable maintenance of the cladding and primary systems are assured, and that violations of the safety limits require plant shutdown and regulatory review. The deletion also maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A26 The "Limiting Safety System Settings (LSSS)" definition is deleted because the definition already exists in 10 CFR 50.36 and does not need repeating in the Technical Specifications. The deletion of this definition also maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A27 The "Limiting Conditions for Operation (LCO)" definition is deleted because the definition already exists in 10 CFR 50.36 and does not need to be repeated in the Technical Specifications. Each proposed LCO clearly depicts the minimum acceptable levels of system performance required to assure safe startup and operation of the facility. The deletion of this definition also maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A28 The "PRIOR TO STARTUP" definition is deleted because it will not be used in the BFN ISTS. The Applicability, Frequency, and Completion Times in the BFN ISTS contain specific plant operation modes and do not need further clarification. The deletion of this definition also maintains the consistency between BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A29 The "Operating" definition is deleted because this state of a system does not need to be explicitly defined when considering whether or not the design function can be met. Whether a system is "Operating" or "shut down" does not provide relief concerning Operability requirements. The definition of Operable or Operability is sufficient in this case.



JUSTIFICATION FOR CHANGES
SECTION 1.0 - USE AND APPLICATION

ADMINISTRATIVE CHANGES (CONTINUED)

- Operability is assumed until the system, etc. is found to be inoperable by failure anytime or during the performance of the Surveillance Requirements at the specified frequencies. The deletion of this definition also maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A30 The definition of "Immediate" is deleted because it will not be used in the BFN ISTS. The term "Immediately," however, will be used. The use of this term is defined in Section 1.3 of the BFN ISTS. The deletion of this definition also maintains the consistency between the BFN ISTS and the BWR/4 ISTS. This is an administrative change because the term is being moved from one section to another. The removal of this definition is considered administrative with no impact of its own.
- A31 The definition of "HOT STANDBY CONDITION" is deleted because it is no longer needed. The Startup Mode or Mode 2 contains the conditions of Hot Standby (< 1% power) but does not encompass the intent of the Hot Standby Condition. The intent of Hot Standby Condition is to be reducing power and not increasing power as in the Startup Mode. Hot Standby is a condition that is often hard to maintain for many plants and its use was phased out in later BWR operating plants. Actions will require the plant to be placed in an appropriate Mode or other condition instead of the Hot Standby Condition. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A32 The definition of "COLD CONDITION" is deleted. Creation of Table 1.1-1 on reactor modes includes appropriate definitions for modes based on reactor mode switch position, reactor coolant temperatures, and reactor head bolt tensioning. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A33 The definition of "SHUTDOWN CONDITION" is deleted. Creation of Table 1.1-1 on reactor modes includes appropriate definitions for modes based on reactor mode switch position, reactor coolant temperatures, and reactor head bolt tensioning. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A34 The definition of "Modes of Operation" was deleted because it will not be used in the LCOs or Surveillance Requirements of the BFN ISTS. The Modes of Operation definition was incorporated into Modes table through the definitions of the positions of the mode switch which determines the



JUSTIFICATION FOR CHANGES
SECTION 1.0 - USE AND APPLICATION

ADMINISTRATIVE CHANGES (CONTINUED)

- mode of operation. Terms not used in the LCOs or Surveillance Requirements of the Technical Specifications do not need to be defined. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A35 The definition of "Primary Containment Integrity" is deleted because of the confusion associated with this definition compared to its use in its respective LCO. All the requirements are specifically addressed in the LCO along with other LCOs in the Containment Systems Section (Section 3.6). The Bases for these LCOs also contain a description of what constitutes primary containment. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A36 The definition of "Secondary Containment Integrity" is deleted because of the confusion associated with this definition compared to its use in its respective LCO. All the requirements are specifically addressed in the LCO along with other LCOs in the Containment Systems Section (Section 3.6). The Bases for these LCOs also contain a description of what constitutes secondary containment. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A37 The definition of "Operating Cycle" is deleted because it is not used in the LCOs or Surveillance Requirements of the BFN ISTS. The BFN ISTS uses specific times (18 months instead of every Refueling) as designations for Surveillance Frequencies. Terms not used in the LCOs or Surveillance Requirements of the Technical Specifications do not need to be defined. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A38 The definition of "Refueling Outage" is deleted because it is not used in the LCOs or Surveillance Requirements of the BFN ISTS. The BFN ISTS uses specific times (18 months instead of every Refueling) as designations for Surveillance Frequencies. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.

JUSTIFICATION FOR CHANGES
SECTION 1.0 - USE AND APPLICATION

ADMINISTRATIVE CHANGES (CONTINUED)

- A39 The definition of "Reactor Vessel Pressure" (the term "Reactor Pressure Vessel Pressure" is used in the BWR/4 ISTS) was deleted because it is clearly depicted in the BFN ISTS Bases that it is the pressure measured by the steam dome detectors (Ref. Section 3.4.10). The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A40 The definition of "Transition Boiling" is deleted because it is not used in either the LCOs or Surveillance Requirements. A discussion of MCPR and transition boiling is found in Bases 3.2.2. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A41 The definitions of the following terms are deleted because they are not used in either the LCOs or Surveillance Requirements of the BFN ISTS.

Channel
Instrument Functional Test
Source Check (Unit 2 only)
Simulated Automatic Actuation
Instrument Calibration

Trip System
Protective Action
Protective Function
Logic
Core Maximum Fraction of Critical
Power

Some of these terms are encompassed in the definitions of "Channel Check," "Channel Functional Test," and "Channel Calibration." Any changes to Surveillance Requirements are justified in the appropriate Technical Specifications Section change package. The deletion of these definitions maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of these definitions is considered administrative with no impact of its own.

- A42 The definition of "Functional Test" is deleted because it is not used in either the LCOs or Surveillance Requirements. The definition of "Functional Test" is "the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water)." These types of tests in the BFN ISTS are called out directly in the Surveillance Requirements (e.g., verify the following ECCS pumps develop the specified flow rate). Post maintenance functional testing is covered by plant procedure and not usually in the Technical Specifications. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.

JUSTIFICATION FOR CHANGES
SECTION 1.0 - USE AND APPLICATION

ADMINISTRATIVE CHANGES (CONTINUED)

- A43 The definition of "Shutdown" is deleted. A "MODES" table that definitively defines separate modes and eliminates the possibility of being in more than one Mode at a time has been created (proposed Table 1.1-1). The Shutdown condition can be either in Mode 3, 4, or 5. This is made clear in the Modes table which shows other criteria that must be met to be in those Modes. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A44 The definition of "Engineered Safeguards" was deleted because it is not used in either the LCOs or Surveillance Requirements of the BFN ISTS. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A45 The definition of "Reportable Event" was deleted because it is not used in either the LCOs or Surveillance Requirements of the BFN ISTS. The use of Reportable Event is covered in 10 CFR 50.73 and does not need to be defined in the Technical Specifications. Review of Reportable Events is covered in Section 5.0 of the Technical Specifications. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A46 The definition of "Purge or purging" was deleted because it is not used in either the LCOs or Surveillance Requirements of the BFN ISTS. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A47 The definition of "Unrestricted Area" was deleted because it is not used in the LCOs or Surveillance Requirements of the BFN ISTS. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A48 The definition of "Gaseous Waste Treatment System" was deleted because it is not used in the LCOs or Surveillance Requirements of the BFN ISTS. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.



JUSTIFICATION FOR CHANGES
SECTION 1.0 - USE AND APPLICATION

ADMINISTRATIVE CHANGES (CONTINUED)

- A49 The definition of "Members of the Public" was deleted because it is not used in the LCOs or Surveillance Requirements of the BFN ISTS. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A50 The definition of "Limiting Control Rod Pattern" was deleted because it is not used in the LCOs or Surveillance Requirements of the BFN ISTS. The deletion of this definition maintains the consistency between the BFN ISTS and the BWR/4 ISTS. The removal of this definition is considered administrative with no impact of its own.
- A51 The Surveillance Frequency Notation Table is being deleted because the Surveillance Requirement Frequencies in the BFN ISTS do not use notation. The Frequencies are specific by giving the number of hours, days, or months (e.g., instead of M the BFN ISTS will have 31 days). The change of the "Daily" SI frequency from once per normal calendar day (midnight to midnight) to once per 24 hours has been categorized as administrative since over the long-term the frequencies will be about the same. However, the change could be considered more restrictive and less restrictive. It is more restrictive since CTS would allow up to approximately 48 hours (12:01 am on one day to 11:59 pm on the next day). However, the next day you could only go to 11:59 PM and would likely go less and the previous day you would have likely gone less than 24 hours. It is less restrictive since the new Frequency would always allow as long as 30 hours (24 + 25%). Over the long-term, on the average the CTS and ISTS frequency will be about the same.
- A52 The rewording and title capitalization of the "Core Maximum Fraction of Limiting Power Density" definition is administrative because the meaning was not changed. During the BWR/4 ISTS development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. This change maintains the consistency between the BFN ISTS and the BWR/4 ISTS.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 Not Used.
- M2 The proposed Startup Mode will now include the mode switch position of "Refuel" when the head bolts are fully tensioned (footnote (a)). The change eliminates the potential to interpret certain plant conditions such that no MODE, or a less restrictive MODE, would exist. Currently, CTS 1.0.M allows the plant to be considered in the SHUTDOWN CONDITION and



JUSTIFICATION FOR CHANGES
SECTION 1.0 - USE AND APPLICATION

TECHNICAL CHANGES - MORE RESTRICTIVE (CONTINUED)

in the Shutdown Mode with the mode switch in the Refuel position (and other positions are allowed while in the Shutdown Mode) as permitted by notes to that definition. By defining this plant condition as STARTUP MODE, sufficiently conservative restrictions will be applied by applicable LCOs.

The allowance to place the Mode Switch in other positions has been moved to Section 3.10, Special Operations and Section 3.3.2.1, Control Rod Block Instrumentation. Any technical changes to these allowances will be discussed in the Justification for Changes to these Sections.

TECHNICAL CHANGES - LESS RESTRICTIVE

- L1 The words "or actual" in reference to the injected signal have been added to the definition of CHANNEL FUNCTIONAL TEST. Some CHANNEL FUNCTIONAL TESTS are being performed by insertion of the actual signal into the logic (e.g., rod block interlocks). For others, there is no reason why an actual signal would preclude satisfactory performance of the test. Use of an actual signal instead of the existing requirement, which limits use to a simulated signal, will not affect the performance of the channel. OPERABILITY can be adequately demonstrated in either case since the channel itself cannot discriminate between "actual" or "simulated."
- L2 Since the requirements are essentially the same, the analog and bistable channel requirements have been combined. The only technical difference is the location of the injected signal. As provided in the definition of CHANNEL FUNCTIONAL TEST for instruments with analog channels, the signal used to test bistable channels is proposed to be allowed to be injected "as close to the sensor as practicable." Injecting a signal at the sensor would in some cases involve significantly increased probabilities of initiating undesired circuits during the test since several logic channels are often associated with a particular sensor. Performing the test by injection of a signal at the sensor requires jumpering of the other logic channels to prevent their initiation during the test, or increases the scope of the test to include multiple tests of the other logic channels. Either method significantly increases the difficulty of performing the surveillance. Allowing initiation of the signal close to the sensor provides a complete test of the logic channel while significantly reducing the probability of undesired initiation. In addition, the sensor is still being checked during a CHANNEL CALIBRATION.



JUSTIFICATION FOR CHANGES
SECTION 1.0 - USE AND APPLICATION

TECHNICAL CHANGES - LESS RESTRICTIVE (CONTINUED)

- L3 This change is proposed to allow control rod movement in a defueled cell to not be considered a CORE ALTERATION. In this configuration, the negative reactivity inserted by removing the adjacent four fuel assemblies is significantly more than any minimal positive reactivity inserted during the removal of the control rod. Appropriate Technical Specification controls are applied during the fuel movements preceding the control rod removal to protect from or mitigate a reactivity excursion event. After the fuel has been removed, sufficient margin and design features (the design of a control rod precludes its removal without all fuel assemblies in the cell removed) are in place to allow removing the Technical Specification controls during the control rod removal. The proposed change focuses the definition on activities that can affect core reactivity. Maintaining CORE ALTERATIONS as movement of only that which can affect core reactivity is consistent with the BWR Standard Technical Specifications, NUREG 1433. The basis for this is evident in that the Specifications that are applicable during CORE ALTERATIONS are those that protect from or mitigate a reactivity excursion event.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



1.1/2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

(A1)

(A1)

~~Specifications~~

2.0 Safety Limits (SLs)

2.1 SLs

Reactor Core SLs

2.1.1 ~~A. Thermal Power Limits~~

with H/C Steam done

2.1.1.2 1. A Reactor Pressure, 800 psia and Core flow $\geq 10\%$ of Rated Core flow: ≥ 2785

(A3)

(A2)

except as marked

(A3)

When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.10 shall constitute violation of the fuel cladding integrity safety limit. MCPR shall be ≥ 1.10

Specifications

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode) (Flow biased)

a. When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

See Justification for Changes to BFN ISTS 3.3.1.1



~~1.1/2.1 FUEL CLADDING INTEGRITY~~

~~LIMITING SAFETY SYSTEM SETTING~~

A1

2.0 SAFETY LIMIT (SLs)

~~1.1.A Thermal Power Limits~~

2.1.A Neutron Flux Trip Settings

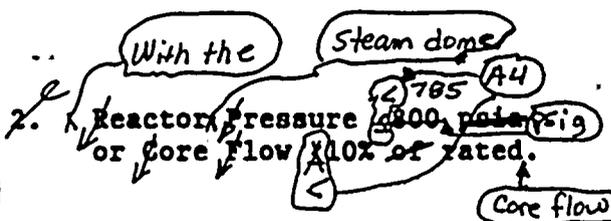
- d. Fixed High Neutron Flux Scram Trip Setting—When the mode switch is in the RUN position, the APRM fixed high flux scram trip setting shall be:

≤ 120% power.

- 2. APRM and IRM Trip Settings (Startup and Hot Standby Modes).

- a. APRM—When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power.
- b. IRM—The IRM scram shall be set at less than or equal to 120/125 of full scale.

2.1.1.1



When the reactor pressure is ≤ 800 psia or core flow is $\le 10\%$ of rated, the core thermal power shall not exceed 823 MWt (.25% of rated thermal power).

Thermal Power shall be $\le 25\%$ RTP.

A3

Except as marked

See Justification for Changes to BFN ISTS 3.3.1.1



(A1) 1.1/2.1 FUEL CLADDING INTEGRITY e

SAFETY LIMIT (SLs)

1.1.B. Power Transient

To ensure that the SAFETY LIMITS established in Specification 1.1.A are not exceeded, each required scram shall be initiated by its expected scram signal. The SAFETY LIMIT shall be assumed to be exceeded when scram is accomplished by means other than the expected scram signal.

(L1)

C. Reactor Vessel Water Level

Whenever there is irradiated fuel in the reactor vessel, the water level shall be greater than or equal to 372.5 inches above vessel zero.

(A3)

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

(L2)

LIMITING SAFETY SYSTEM SETTING

2.1.B. Power Transient Trip Settings

1. Scram and isolation (PCIS groups 2,3,6) reactor low water level \geq 538 in. above vessel zero
2. Scram—turbine stop valve closure \leq 10 percent valve closure
3. Scram—turbine control valve fast closure or turbine trip \geq 550 psig
4. (Deleted)
5. Scram—main steam line isolation \leq 10 percent valve closure
6. Main steam isolation valve closure—nuclear system low pressure \geq 825 psig

C. Water Level Trip Settings

1. Core spray and LPCI actuation—reactor low water level \geq 398 in. above vessel zero
2. HPCI and RCIC actuation—reactor low water level \geq 470 in. above vessel zero
3. Main steam isolation valve closure—reactor low water level \geq 398 in. above vessel zero

See Justification for Changes to BFN ISTS 3.3.1.1, 3.3.5.1, 3.3.5.2 and 3.3.6.1

1.2/2.2 REACTOR COOLANT SYSTEM INTEGRITY

SAFETY LIMIT (SLs)

1.2 Reactor Coolant System Integrity

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

A1

LIMITING SAFETY SYSTEM SETTING

2/2 Reactor Coolant System Integrity

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

A3

Specifications

A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

SL 2.1.2

Reactor Steam dome pressure shall be ≤ 1325 psig

See Justification for Changes to BFN ISTS 3.3.1.1 and 3.4.3

Specifications

The limiting safety system settings shall be as specified below:

<u>Protective Action</u>	<u>Limiting Safety System Setting</u>
A. Nuclear system relief valves open--nuclear system pressure	1,105 psig \pm 11 psi (4 valves)
	1,115 psig \pm 11 psi (4 valves)
	1,125 psig \pm 11 psi (5 valves)
B. Scram--nuclear system high pressure	$\leq 1,055$ psig

6.3 PLANT STAFF QUALIFICATIONS

Each member of the unit's staff shall meet or exceed the minimum qualifications for comparable positions as specified in the TVA Nuclear Quality Assurance Plan (TVA-NQA-PLN89-A).

6.4 (Deleted)

6.5 (Deleted)

6.6 (Deleted)

see Justification for Changes
for BFN 15TS 5.0

2.2

~~6.7~~ SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

- a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Site Vice President and the NSRB shall be notified within 24 hours.
- b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. 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.
- c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB, and the Site Vice President within 14 days of the violation.

(M1) Proposed 2.2 →

4-LA1



AI 6.7.1 (Cont'd)

LAI

d. ~~Critical operation of the unit shall not be resumed until authorized by the Commission.~~

6.8 PROCEDURES/INSTRUCTIONS AND PROGRAMS

See Justification for Changes for BFN ISTS 5.0

6.8.1 PROCEDURES

6.8.1.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

- a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.
- b. Limitations on the amount of overtime worked by individuals performing safety-related functions in accordance with NRC Policy statement on working hours (Generic Letter No. 82-12).
- c. Surveillance and test activities of safety-related equipment.
- d. (Deleted)
- e. (Deleted)
- f. Fire Protection Program implementation.
- g. (Deleted)
- h. (Deleted)
- i. Offsite Dose Calculation Manual.
- j. Administrative procedures which control technical and cross-disciplinary review.



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



1.1/2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

(A1)

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

(A1) ~~Specifications~~

2.0 SAFETY LIMITS (SLs)

2.1 SLs

Reactor Core SLs

2.1.1 ~~Thermal Power Limits~~

With the Steamdome

2.1.1.2 ~~Reactor Pressure, 800 psia and Core Flow ≥ 785~~

$\geq 10\%$ of Rated Core flow

(A3)

except as marked

When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.10 shall constitute violation of the fuel cladding integrity safety limit.

(A3)

MCPR shall be ≥ 1.10

Specifications

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (RUN Mode) (Flow Biased)

a. When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

See justification for changes to BFN 1STS 3.3.1.1



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~~1.1/2.1 FUEL CLADDING INTEGRITY~~

(A1) 2.0 SAFETY LIMITS (SLs)

~~LIMITING SAFETY SYSTEM SETTING~~

~~1.1.A Thermal Power Limits~~

2.1.A Neutron Flux Trip Settings (Cont'd)

d. Fixed High Neutron Flux Scram Trip Setting—When the mode switch is in the RUN position, the APRM fixed high flux scram trip setting shall be:

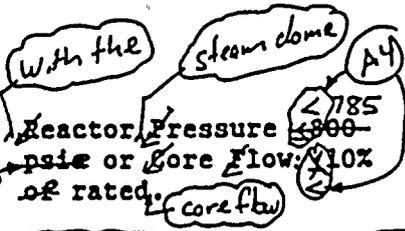
$\leq 120\%$ power.

2. APRM and IRM Trip Settings (Startup and Hot Standby Modes).

a. APRM—When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power.

b. IRM—The IRM scram shall be set at less than or equal to 120/125 of full scale.

2.1.1.1 ~~2. Reactor Pressure ≤ 800 psia or Core Flow $\leq 10\%$ of rated.~~



(A3)
 {except as marked}

When the reactor pressure is ≤ 800 psia or core flow is $\leq 10\%$ of rated, the core thermal power shall not exceed 823 MWt (25% of rated thermal power).

THERMAL POWER shall be $\leq 25\%$ RTP.

{See Justification for Changes to BFN 1STS 3.3.1.1}



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SAFETY LIMITS (SLs)

LIMITING SAFETY SYSTEM SETTING

1.1.B. ~~Power Transient~~

To ensure that the Safety Limits established in Specification 1.1.A are not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by means other than the expected scram signal.

L1

~~G. Reactor Vessel Water Level~~

Whenever there is irradiated fuel in the reactor vessel, the water level shall be greater than or equal to 372.5 inches above vessel zero.

A3

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

L2

2.1.B. ~~Power Transient Trip Settings~~

- 1. Scram and isolation (PCIS groups 2,3,6) reactor low water level \geq 538 in. above vessel zero
- 2. Scram—turbine stop valve closure \leq 10 percent valve closure
- 3. Scram—turbine control valve fast closure or turbine trip \geq 550 psig
- 4. (Deleted)
- 5. Scram—main steam line isolation \leq 10 percent valve closure
- 6. Main steam isolation valve closure—nuclear system low pressure \geq 825 psig

C. ~~Water Level Trip Settings~~

- 1. Core spray and LPCI actuation— reactor low water level \geq 398 in. above vessel zero
- 2. HPCI and RCIC actuation— reactor low water level \geq 470 in. above vessel zero
- 3. Main steam isolation valve closure— reactor low water level \geq 398 in. above vessel zero

See Justification for Changes to BFN ISTS 3.3.1.1, 3.3.5.1, 3.3.5.2, and 3.3.6.1

1.2/2.2 REACTOR COOLANT SYSTEM INTEGRITY

SAFETY LIMITS (SLs)

LIMITING SAFETY SYSTEM SETTING

1.2 Reactor Coolant System Integrity

2.2 Reactor Coolant System Integrity

Applicability

Applicability

Applies to limits on reactor coolant system pressure.

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

A1

Specifications

Specifications

A3

The pressure at the lowest point of the reactor vessel shall not exceed 1.375 psig whenever irradiated fuel is in the reactor vessel.

The limiting safety system settings shall be as specified below:

SL 2.1.2

Reactor steam dome pressure shall be ≤ 1325 psig.

Limiting Safety Protective Action System Setting

A. Nuclear system relief valves open--nuclear system pressure	1,105 psig \pm 11 psi (4 valves)
	1,115 psig \pm 11 psi (4 valves)
	1,125 psig \pm 11 psi (5 valves)
B. Scram- nuclear system high pressure	$\leq 1,055$ psig

See Justification for Changes to BFN ISTS 3.3.1.1 and 3.4.3



6.3 PLANT STAFF QUALIFICATIONS

Each member of the unit's staff shall meet or exceed the minimum qualifications for comparable positions as specified in the TVA Nuclear Quality Assurance Plan (TVA-NQA-PLN89-A).

6.4 (Deleted)

See Justification for Changes
for BFN 1STS 5.0

6.5 (Deleted)

6.6 (Deleted)

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

- a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Site Vice President and the NSRB shall be notified within 24 hours.
- b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. 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.
- c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB, and the Site Vice President within 14 days of the violation.
- d. Critical operation of the unit shall not be resumed until authorized by the Commission.

(M1) Proposed 2.2

6.0-5

PAGE 6 OF 7

6.8 PROCEDURES/INSTRUCTIONS AND PROGRAMS

6.8.1 PROCEDURES

See Justification for
Changes to BFN ISTS 5.0

6.8.1.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

- a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.
- b. Limitations on the amount of overtime worked by individuals performing safety-related functions in accordance with NRC Policy statement on working hours (Generic Letter No. 82-12).
- c. Surveillance and test activities of safety-related equipment.
- d. (Deleted)
- e. (Deleted)
- f. Fire Protection Program implementation.
- g. (Deleted)
- h. (Deleted)
- i. Offsite Dose Calculation Manual.
- j. Administrative procedures which control technical and cross-disciplinary review.



UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



1.1/2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

(A1)

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

(A1)

Specification

2.0 SAFETY LIMITS (SLs)

2.1 SLs

Reactor Core SLs

2.1.1 Thermal Power Limits

with the steam dome (A2)

2.1.1.2

Reactor pressure ≥ 800 psia and core flow $\geq 10\%$ of Rated (A2)

(A3)

except as marked

When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.10 shall constitute violation of the fuel cladding integrity safety limit. MCPR shall be ≥ 1.10 (A3)

Specification

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode) (Flow Biased)

a. When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

See justification for changes to BFN ISTS 3.3.1.1



~~1.1/2.1 FUEL CLADDING INTEGRITY~~

2.0 SAFETY LIMITS (SL)

~~LIMITING SAFETY SYSTEM SETTINGS~~

~~1.1.A Thermal Power Limits~~

2.1.A Neutron Flux Trip Settings

- d. Fixed High Neutron Flux Scram Trip Setting—When the mode switch is in the RUN position, the APERM fixed high flux scram trip setting shall be:

$S \leq 120\%$ power.

- 2. APERM and IRM Trip Settings (Startup and Hot Standby Modes).

- a. APERM—When the reactor mode switch is in the STARTUP position, the APERM scram shall be set at less than or equal to 15% of rated power.
- b. IRM—The IRM scram shall be set at less than or equal to 120/125 of full scale.

2.1.1.1 2.

With the Steam dome A4
 Reactor Pressure ≤ 785 psia, psig
 or Core Flow $\leq 10\%$ of rated. Core flow

When the reactor pressure is ≤ 800 psia or core flow is $\leq 10\%$ of rated, the core thermal power shall not exceed 823 Mwt (.25% of rated thermal power).

Thermal Power shall be $\leq 25\%$ RTP

A3

except as marked

See Justification for Changes to BFN ISTS 3.3.1.1



(A1)

SAFETY LIMIT (SLs)

1.1.B. Power Transient

To ensure that the SAFETY LIMITS established in Specification 1.1.A are not exceeded, each required scram shall be initiated by its expected scram signal. The SAFETY LIMIT shall be assumed to be exceeded when scram is accomplished by means other than the expected scram signal.

(L1)

~~C. Reactor Vessel Water Level~~

Whenever there is irradiated fuel in the reactor vessel, the water level shall be greater than or equal to 372.5 inches above vessel zero.

(A3)

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

(L2)

LIMITING SAFETY SYSTEM SETTING

2.1.B. Power Transient Trip Settings

- 1. Scram and isolation (PGIS groups 2,3,6) reactor low water level \geq 538 in. above vessel zero
- 2. Scram—turbine stop valve closure \leq 10 percent valve closure
- 3. Scram—turbine control valve fast closure or turbine trip \geq 550 psig
- 4. (Deleted)
- 5. Scram—main steam line isolation \leq 10 percent valve closure
- 6. Main steam isolation valve closure—nuclear system low pressure \geq 825 psig

C. Water Level Trip Settings

- 1. Core spray and LPCI actuation—reactor low water level \geq 398 in. above vessel zero
- 2. HPCI and RCIC actuation—reactor low water level \geq 470 in. above vessel zero
- 3. Main steam isolation valve closure—reactor low water level \geq 398 in. above vessel zero

See Justification for Changes to BFN ISTS 3.3.1.1, 3.3.5.1, 3.3.5.2 and 3.3.6.1



1.2/2.2 REACTOR COOLANT SYSTEM INTEGRITY

SAFETY LIMIT (SLs)

1.2 Reactor Coolant System Integrity

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

A1

LIMITING SAFETY SYSTEM SETTING

2.2 Reactor Coolant System Integrity

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specification

A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

A3

SL 2.1.2

Reactor steam dome pressure shall be ≤ 1325 psig

See Justification for Changes to BFN ISTS 3.3.1.1 and 3.4.3

Specification

The limiting safety system settings shall be as specified below:

- | | |
|--|------------------------------------|
| A. Nuclear system relief valves open—nuclear system pressure | 1,105 psig \pm 11 psi (4 valves) |
| | 1,115 psig \pm 11 psi (4 valves) |
| | 1,125 psig \pm 11 psi (5 valves) |
| B. Scram—nuclear system high pressure | $\leq 1,055$ psig |

6.3 PLANT STAFF QUALIFICATIONS

Each member of the unit's staff shall meet or exceed the minimum qualifications for comparable positions as specified in the TVA Nuclear Quality Assurance Plan (TVA-NQA-PLN89-A).

6.4 (Deleted)

See Justification for Changes
for BFN 1STS 5.0

6.5 (Deleted)

6.6 (Deleted)

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

- a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Site Vice President and the NSRB shall be notified within 24 hours.
- b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. 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.
- c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB, and the Site Vice President within 14 days of the violation.
- d. Critical operation of the unit shall not be resumed, until authorized by the Commission.

MI Proposed 2.2 →

LA1



See Justification for
changes for BFN.ISTS 5.0

6.8 PROCEDURES/INSTRUCTIONS AND PROGRAMS

6.8.1 PROCEDURES

6.8.1.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

- a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.
- b. Limitations on the amount of overtime worked by individuals performing safety-related functions in accordance with NRC Policy statement on working hours (Generic Letter No. 82-12).
- c. Surveillance and test activities of safety-related equipment.
- d. (Deleted) +
- e. (Deleted) +
- f. Fire Protection Program implementation.
- g. (Deleted)
- h. (Deleted) +
- i. Offsite Dose Calculation Manual.
- j. Administrative procedures which control technical and cross-disciplinary review.



JUSTIFICATION FOR CHANGES
SECTION 2.0 - SAFETY LIMITS

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The reactor pressure limit unit of measure has been changed from psia to psig. In addition, the requirement for when the MCPR limit is applicable has been reduced slightly (by adding the "equal to" sign) for consistency with the BWR Standard Technical Specifications, NUREG-1433. The limit on core flow is now specified as greater than or equal to. The current Safety Limits do not address the situation when core flow is equal to the limit. While these changes are actually more restrictive, since they are so minor, they are considered an administrative changes.
- A3 The Safety Limits were reworded without changing the intent of the Safety Limit (no technical changes were made). Editorial rewording is consistent with the BWR Standard Technical Specification, NUREG-1433. During its development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the TS. Units for pressure has been changed from psia to psig. The Safety Limit was changed from pressure (1375) at the lowest point of the reactor vessel to reactor steam dome pressure (1325), which is equivalent considering water level differences.



**JUSTIFICATION FOR CHANGES
SECTION 2.0 - SAFETY LIMITS**

- A4 The "equal to" was taken out of "less than or equal to" symbol. This was done for consistency with the current BFN Bases for the Safety Limit which states that a core thermal power limit of 25 percent for reactor pressures below 800 psia (785 psig) is conservative. This is also consistent with NUREG-1433. Also the "equal to" was taken out of "less than or equal to" symbol as it relates to rated core flow to maintain consistency between the current technical specifications and NUREG-1433.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 A new 2.2 requirement is added to the Safety Limit Violations Section, which requires all SLs to be restored and all insertable rods inserted within 2 hours. Exceeding a Safety Limit may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100 limits. These requirements ensure that the operators take prompt remedial action and also ensure that the probability of an accident occurring when a Safety Limit is violated is minimal.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 BFN proposes the requirements related to Safety Limit Violation reporting and restart authorization not be retained in Technical Specifications. Duplication of the regulations provided in 10 CFR 50.36, 50.72 and 50.73 is not necessary to assure safe operation of the facility. The current regulations require BFN to perform all the actions currently required by Technical Specifications. This change is consistent Technical Specification Change Traveler TSTF-5 (approved by NRC on 11/27/95) and Revision 0 to Generic Change BWROG-09, which addressed several NRC and Industry initiatives to improve the content and presentation of Administrative Controls.



1.0 DEFINITIONS

See Justification for Changes to BFW .1 STS Section 1.0

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

- A. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit requires unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.
- B. Limiting Safety System Settings (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.
- C. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

1. In the event a Limiting Condition for Operation and/or associated requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours unless corrective measures are completed that permit operation under the permissible discovery or until the reactor is placed in an operational condition in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications. This provides actions to be taken for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. For example, if a specification calls for two systems (or subsystems) to be operable and provides for explicit requirements if one system (or subsystem) is inoperable, then if both systems (or subsystems) are inoperable the unit is to be in at least Hot Standby in 6 hours and in Cold Shutdown within the following 30 hours if the inoperable condition is not corrected.

(L1) Replace with Proposal LCO 3.0.3

- (M1) → ADD LCO 3.0.1
- (M2) → ADD LCO 3.0.2
- (M3) → ADD LCO 3.0.4
- (L2) → ADD LCO 3.0.5
- (A2) → ADD LCO 3.0.6
- (A3) → ADD LCO 3.0.7 1.0-1
- (A1) → ADD LCO BASES

BFW Unit 1



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



JUSTIFICATION FOR CHANGES
SECTION 2.0 - SAFETY LIMITS

"Specific"

- L1 The proposed change deletes the "Power Transient" Safety Limit. The intent of this Safety Limit was to ensure that other Safety Limits are not exceeded. This Safety Limit is assumed to be exceeded when a scram is accomplished by means other than the expected scram signal. The scram setpoints are established in order to ensure margin to the safety limits. Exceeding the scram setpoint, in and of itself, does not necessarily indicate that a Safety Limit has been exceeded. Section 2.1.B of the present BFN TS contains six power transient trip settings that initiate a reactor scram. These scram setpoints are included in Table 3.3.1.1-1 of the new ISTS. The surveillance requirements imposed on these scram setpoints in Table 3.3.1.1-1 help to ensure that the margin to a safety limit is preserved. The redundancy built into the RPS system is maintained by the action provisions of ISTS 3.3.1.1. Therefore, the intent of present Power Transient Safety Limit 1.1.B is maintained by the proposed provisions in ISTS 3.3.1.1 for the RPS. Additionally, although the proposed change deletes the requirement for assuming the Safety Limit is exceeded when scram is accomplished by means other than the expected scram signal, the proposed change does not preclude the required actions if the Safety Limit is actually violated.
- L2 The current Safety Limit for the reactor vessel water level is that level shall be maintained not less than 372.5 inches above vessel zero. This proposed Safety Limit is that level should be greater than the top of the active irradiated fuel (approximately 366 inches above vessel zero). This represents a less restrictive change since the top of the active irradiated fuel at BFN Unit 2 is less than 372.5 inches above vessel zero. The change still ensures adequate margin for effective action in the event of a level drop. This change is consistent with NUREG-1433.



See Justification for Changes to BFN 1STS Section 1.0

TS 373 Specification 3.0

1.0 DEFINITIONS (Cont'd)

- GG. Site Boundary - Shall be that line beyond which the land is not owned, leased, or otherwise controlled by TVA.
- HH. Unrestricted Area - 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 industrial, commercial, institutional, or recreational purposes.
- II. Dose Equivalent I-131 - The DOSE EQUIVALENT I-131 shall be the concentration of I-131 (in $\mu\text{Ci/gm}$) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factor used for this calculation shall be those listed in Table III of TID-14844 "Calculation of Distance Factors for Power and Test Reactor Sites".
- JJ. Gaseous Waste Treatment System - The charcoal adsorber vessels installed on the discharge of the steam jet air ejector to provide delay to a unit's offgas activity prior to release.
- KK. Members of the Public - An individual in a controlled or UNRESTRICTED AREA. However, an individual is not a MEMBER OF THE PUBLIC during any period in which the individual receives an occupational dose (as defined in 10 CFR 20).

A4

Replace with Proposed SR 3.0.1

Replace with Proposed SR 3.0.2

A5 M4 L3

LL. Surveillance - Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual limiting conditions for operation unless otherwise stated in an individual Surveillance Requirements. Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval. It is not intended that this (extension) provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages.

Performance of a Surveillance Requirement within the specified time interval shall constitute compliance and OPERABILITY requirements for a limiting condition for operation and associated action statements unless otherwise required by these specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, the 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.

M5 ADD SR 3.0.4

A1 ADD SR Bases



1.0. ~~DEFINITIONS (Cont'd)~~ (A)

SR 3.0.3
(Cont.)

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

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

MM. Surveillance Requirements for ASME Section XI Pump and Valve Program - Surveillance Requirements for Inservice Testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

1. 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 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 50.55(g)(6)(i).
2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice 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 testing activities	Required frequencies for performing inservice 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 Justification for Changes for BFN 15TS 55

3. The provisions of Specification 1.0.LL are applicable to the above required frequencies for performing inservice testing activities.
4. Performance of the above inservice testing activities shall be in addition to other specified surveillance requirements.
5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any technical specification.
6. The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with staff positions on schedule, methods, personnel, and sample expansion included in this generic letter.



UNIT 2

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



See Justification for Changes
to BFN ISTS Section 1.0

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit requires unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

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

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

L1

Replace with
Proposed.
LCO 3.0.3

1. ~~In the event a Limiting Condition for Operation and/or associated requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours unless corrective measures are completed that permit operation under the permissible discovery or until the reactor is placed in an operational condition in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications. This provides actions to be taken for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. For example, if a specification calls for two systems (or subsystems) to be operable and provides for explicit requirements if one system (or subsystem) is inoperable, then if both systems (or subsystems) are inoperable the unit is to be in at least Hot Standby in 6 hours and in Cold Shutdown within the following 30 hours if the inoperable condition is not corrected.~~

- M1 → ADD LCO 3.0.1
- M2 → ADD LCO 3.0.2
- M3 → ADD LCO 3.0.4
- L2 → ADD LCO 3.0.5
- A2 → ADD LCO 3.0.6
- BFN Unit 2 (A3) → ADD LCO 3.0.7 1.0-1
- (A1) → ADD LCO Bases

See Justification for Changes to BFN ISTS 1.0

1.0. DEFINITIONS (Cont'd)

- GG. Site Boundary - Shall be that line beyond which the land is not owned, leased, or otherwise controlled by TVA.
- HH. Unrestricted Area - 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 industrial, commercial, institutional, or recreational purposes.
- II. Dose Equivalent I-131 - The DOSE EQUIVALENT I-131 shall be the concentration of I-131 (in $\mu\text{Ci/gm}$) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factor used for this calculation shall be those listed in Table III of TID-14844 "Calculation of Distance Factors for Power and Test Reactor Sites".
- JJ. Gaseous Waste Treatment System - The charcoal adsorber vessels installed on the discharge of the steam jet air ejector to provide delay to a unit's offgas activity prior to release.
- KK. Members of the Public - An individual in a controlled or UNRESTRICTED AREA. However, an individual is not a MEMBER OF THE PUBLIC during any period in which the individual receives an occupational dose (as defined in 10 CFR 20).

At

Replace with Proposed SR 3.0.1

Replace with Proposed SR 3.0.2

AS M4 L3

LL. Surveillance - Surveillance requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual limiting conditions for operation unless otherwise stated in an individual Surveillance Requirements. Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval. It is not intended that this (extension) provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages.

Performance of a Surveillance Requirement within the specified time interval shall constitute compliance and OPERABILITY requirements for a limiting condition for operation and associated action statements unless otherwise required by these specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

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.

MS Add SR 3.0.4

AI Add SR Bases



1-0 DEFINITIONS (CONT'D) (A1)

SR 3.0.3 (cont.) If the surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.

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

MM. Surveillance Requirements for ASME Section XI Pump and Valve Program
- Surveillance Requirements for Inservice Testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

1. 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 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 50.55(g)(6)(i).
2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these technical specifications:

See Justification for Champs for BFN ISTS 5.5

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required frequencies for performing inservice 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

3. The provisions of Specification 1.0.LL are applicable to the above required frequencies for performing inservice testing activities.
4. Performance of the above inservice testing activities shall be in addition to other specified surveillance requirements.
5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any technical specification.
6. The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, personnel, and sample expansion included in this generic letter.

UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



1.0 DEFINITIONS

See Justification for Changes to BFN ISTS Section 1.0

Specification 3.0

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

- A. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit requires unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.
- B. Limiting Safety System Setting (LSSS) - The limiting safety system setting are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.
- C. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

- 1. In the event a Limiting Condition and/or associated requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours unless corrective measures are completed that permit operation under the permissible discovery or until the reactor is placed in an operational condition in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications. This provides action to be taken for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. For example, if a specification calls for two systems (or subsystems) to be operable and provides for explicit requirements if one system (or subsystem) is inoperable, then if both systems (or subsystems) are inoperable the unit is to be in at least Hot Standby in 6 hours and in Cold Shutdown within the following 30 hours if the inoperable condition is not corrected.

L1 →
Replace with Proposed LCO 3.0.3

- (M1) — ADD LCO 3.0.1
- (M2) — ADD LCO 3.0.2
- (M3) — ADD LCO 3.0.4
- (L2) — ADD LCO 3.0.5
- (A2) — ADD LCO 3.0.6
- (A3) — ADD LCO 3.0.7
- (A1) — ADD LCO Bases.



see justification for changes to BFN ISTS Section 1.0

Specification 3.0
TS 373

1.0 DEFINITIONS (Cont'd)

- GG. Site Boundary - Shall be that line beyond which the land is not owned, leased, or otherwise controlled by TVA.
- HH. Unrestricted Area - 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 industrial, commercial, institutional, or recreational purposes.
- II. Dose Equivalent I-131 - The DOSE EQUIVALENT I-131 shall be the concentration of I-131 (in $\mu\text{Ci/gm}$) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factor used for this calculation shall be those listed in Table III of TID-14844 "Calculation of Distance Factors for Power and Test Reactor Sites".
- JJ. Gaseous Waste Treatment System - The charcoal adsorber vessels installed on the discharge of the steam jet air ejector to provide delay to a unit's offgas activity prior to release.
- KK. Members of the Public - An individual in a controlled or UNRESTRICTED AREA. However, an individual is not a MEMBER OF THE PUBLIC during any period in which the individual receives an occupational dose (as defined in 10 CFR 20).

LL. Surveillance - Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual limiting conditions for operation unless otherwise stated in an individual Surveillance Requirements. Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval. It is not intended that this (extension) provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages.

A4

Replace with Proposed SR 3.0.1

Replace with Proposed SR 3.0.2

A5

M4

L3

Performance of a Surveillance Requirement within the specified time interval shall constitute compliance and OPERABILITY requirements for a limiting condition for operation and associated action statements unless otherwise required by these specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, the 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.

(M5) → Add SR 3.0.4]

(A1) → Add SR Bases]



(A1)
1.0 DEFINITIONS (Cont'd)

SR 3.0.3
(cont.)

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

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

MM. Surveillance Requirements for ASME Section XI Pump and Valve Program - Surveillance Requirements for Inservice Testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

1. 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).
2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice 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 testing activities	Required frequencies for performing inservice 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 Justification for Changes for BFN ISTSS

3. The provisions of Specification 1.0.LL are applicable to the above required frequencies for performing inservice testing activities.
4. Performance of the above inservice testing activities shall be in addition to other specified surveillance requirements.
5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any technical specification.
6. The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, personnel, and sample expansion included in this generic letter.

JUSTIFICATION FOR CHANGES
SECTION 3.0 - LCO APPLICABILITY

ADMINISTRATIVE CHANGES

- A1 The Bases for Section 3.0 are being added in accordance with applicable Bases from NUREG-1433. The individual changes made to add LCO 3.0 and SR 3.0 Technical Specification provisions provide the justifications necessary to substantiate the contents of these Bases.
- A2 LCO 3.0.6 contains new provisions over present Technical Specification requirements. These new provisions provide guidance regarding the appropriate actions to be taken when a single inoperability (e.g., a support system) also results in the inoperability of one or more related systems (e.g., supported system(s)). In the existing Technical Specifications, along with their intent and interpretation provided by the NRC over the years, there is not an unambiguous approach to the combined support/supported system inoperability.
- Guidance provided in the June 13, 1979, NRC memorandum from Brian K. Grimes (Assistant Director for Field Coordination) would indicate an intent/interpretation consistent with the proposed LCO 3.0.6 - without the necessity of also requiring the additional actions of a Safety Function Determination Program. That is, only the inoperable support system actions need be taken.
 - Guidance provided by the NRC in their April 10, 1980, letter to all Licensees regarding the definition of operability and the impact of a support system on the remainder of the Technical Specifications, would indicate a similar philosophy of not taking actions for the inoperable supported equipment. However, in this case, additional actions similar to the proposed Safety Function Determination Program actions, were addressed and required.
 - Generic Letter 91-18 and a plain-English reading of the existing STS provide an interpretation that failure to perform a required function, even as a result of a Technical Specification support system, requires all associated action be taken.

Considering the history of disagreement and misunderstandings in this area, the BWR Standard Technical Specification, NUREG-1433, was developed with industry input and approval of the NRC to include LCO 3.0.6. The present custom BFN Technical Specifications do not contain specific Action Statements as presented in current STS plants. The present BFN Technical Specification provisions are usually left to interpretation as to supported system operability when a support system is inoperable unless specific direction is contained in the Technical Specification. Since the addition of LCO 3.0.6 to the BFN Technical Specifications will clarify existing ambiguities and maintain actions within the realm of previous interpretations, this new provision is deemed to be administrative in nature.



JUSTIFICATION FOR CHANGES
SECTION 3.0 - LCO APPLICABILITY

ADMINISTRATIVE CHANGES
(Continued)

- A3 LCO 3.0.7 is a new requirement that is not presently addressed in the BFN TSs. Present BFN TSs do not contain special operations LCOs and, as such, do not need LCO 3.0.7 allowances. LCO 3.0.7 specifies that compliance with special operations LCOs is optional, but if used, then the actions of the special operations LCOs shall be met. The applicability of a special operations LCO represents a condition not necessarily in compliance with the normal requirements of the TS. A special operation may be performed either under the provisions of the appropriate special operations LCO or under the other applicable TS requirements. When a special operations LCO requires another LCO to be met, only the requirements of the LCO statement are required to be met regardless of that LCOs applicability. The surveillances of the other LCO are not required to be met, unless specified in the Special Operations LCO. This Specification eliminates the confusion which would otherwise exist as to which LCOs apply during the performance of a special test or operation. This change is considered to provide administrative controls for the use of Special Operations LCOs and, as such, this change is administrative in nature.
- A4 Editorial rewording, reformatting and renumbering is made consistent with the BWR Standard Technical Specifications, NUREG-1433. During the development of SR 3.0.1, certain wording preferences or language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications.
- The second sentence of SR 3.0.1 has been worded to clarify an existing intent that is not explicitly stated. The provision states that failure to meet a Surveillance Requirement can occur during performance of the SR or between performances of the SR and either case shall result in failure to meet the LCO.
 - The last sentence of SR 3.0.1 is clarified by adding the phrase "or variables outside specified limits." This addition is necessary since all LCOs do not deal exclusively with equipment operability.
- A5 Editorial rewording for SR 3.0.2 is made consistent with the BWR Standard Technical Specifications, NUREG-1433, which has resulted in no technical changes (either actual or interpretational) to the basic application of the 25% extension to routine surveillances. Also, the sentence "Exceptions to these requirements are stated in the individual Specifications" is added to acknowledge the explicit use of exceptions in various surveillances. These changes provide consistency of wording and clarity for understanding. No technical changes (either actual or interpretational) to the Technical Specifications are intended by these changes.

**JUSTIFICATION FOR CHANGES
SECTION 3.0 - LCO APPLICABILITY**

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 Present BFN Technical Specifications do not contain specific guidance such as proposed in LCO 3.0.1. Present individual Technical Specification requirements are understood to apply as stated in each individual specification without the necessity of a motherhood statement. The adoption of NUREG-1433 for BFN will add requirements not presently in the Technical Specifications and, as such, potentially more restrictive provisions. Proposed LCO 3.0.1 contains exceptions to LCO 3.0.2 and LCO 3.0.7. These exceptions will state when it is acceptable for the LCO to not be met during the Modes or other specified conditions in the Applicability. The exception to LCO 3.0.2 is necessary since LCO 3.0.2 addresses the condition of meeting the associated Actions when not meeting a LCO. The exception to LCO 3.0.7 is necessary since LCO 3.0.7 provides special guidance that will allow Special Operations Technical Specifications to govern over the LCOs in Sections 3.1 through 3.9.
- M2 Present BFN Technical Specifications do not contain the provisions of LCO 3.0.2. LCO 3.0.2 requires that upon failure to meet an LCO, that the required Actions shall be met. Present BFN custom Technical Specifications do not contain a specific Action Statement section. The Actions for the present BFN Technical Specifications are contained within paragraphs that also contain the LCO and Applicability. Exceptions to LCO 3.0.2 are as provided in LCO 3.0.5 and LCO 3.0.6. These exceptions are necessary to allow systems to be returned to service under administrative control to perform the testing required to demonstrate operability per LCO 3.0.5, and to allow a supported system to be considered operable and its Actions not to be taken, solely due to a support system inoperability per LCO 3.0.6. LCO 3.0.2 also contains a provision such that completion of the Required Actions is not required, unless otherwise stated, if the provisions of a LCO are met or are no longer applicable. Since the present BFN TSs do not contain provisions similar to LCO 3.0.2, this proposed change may result in potentially more restrictive requirements.
- M3 The present BFN TSs do not contain provisions similar to proposed LCO 3.0.4. LCO 3.0.4 prohibits entry into a mode or other specified condition, when a LCO is not met, unless the associated actions to be entered permit continued operation for an unlimited period of time. The provisions of LCO 3.0.4 will 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 will not prevent changes in modes or other specified conditions in the applicability that result from a normal shutdown. Any exceptions to this specification are stated in individual specifications. The addition of this specification to BFN TSs represents a potentially more restrictive change.
- M4 The sentence "For Frequencies specified as "once," the above interval extension does not apply" is proposed to be added. The interval extension concept is based on scheduling flexibility for repetitive performances, and these "once" surveillances are not repetitive in

JUSTIFICATION FOR CHANGES
SECTION 3.0 - LCO APPLICABILITY

TECHNICAL CHANGE - MORE RESTRICTIVE (continued)

nature and essentially have no interval as measured from the previous performance. The nature of these SRs precludes the ability to extend their performances, and is therefore a more restrictive requirement. The existing specification can be interpreted to allow the extension to apply to all surveillances.

- M5 Present BFN TSs do not contain provisions similar to proposed 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. This specification applies to changes in modes or other specified conditions in the applicability associated with unit shutdown as well as startup. The provisions of SR 3.0.4 will not prevent changes in modes or other specified conditions in the applicability that are required to comply with actions.

TECHNICAL CHANGE - LESS RESTRICTIVE

- L1 Present Definition 1.0.C.1 contains the requirements typically referred to as LCO 3.0.3 in the Standard Technical Specifications. The changes to present requirements include the following:
- One additional hour for plant shutdown is included over present provisions. This additional hour was included in the STS to allow time to initiate a plant shutdown. The additional hour is included in the ISTS to allow initiation of the plant shutdown and is included in the ISTS times allowed to Mode 2 of 7 hours, to Mode 3 of 13 hours, and to Mode 4 of 37 hours.
 - Present provisions require the unit to be in Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours. The ISTS requires the unit to be in Mode 2 (Startup/Hot Standby) within 7 hours, Mode 3 (Hot Shutdown) within 13 hours and Mode 4 (Cold Shutdown) within 37 hours. This change will relax present shutdown provisions since Mode 2 can be entered at less than approximately 15% rated thermal power whereas present requirements to be in Hot Standby would require the unit to be less than or equal to 1% rated thermal power within 6 hours. This is offset by a more restrictive requirement to be in Mode 3 in 13 hours.
- L2 Present BFN Technical Specifications do not contain the provisions of LCO 3.0.5 from NUREG-1433. 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 Specifications actions require an



JUSTIFICATION FOR CHANGES
SECTION 3.0 - LCO APPLICABILITY

TECHNICAL CHANGE - LESS RESTRICTIVE (continued)

inoperable component to be removed from service, such as maintaining an isolation valve closed, disarming a control rod, or tripping an inoperable instrument channel. To allow performance of Surveillance Requirements to demonstrate the operability of the equipment being returned to service, or to demonstrate the operability of other equipment 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 present Technical Specifications. Without this allowance certain components could not be restored to operable status and a plant shutdown would ensue. Clearly, this 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.

- L3 The sentence "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 proposed to be added. 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.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



~~3.3/4.3~~ REACTIVITY CONTROL SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.3 REACTIVITY CONTROL

Applicability

Applies to the operational status of the control rod system.

Objective

To assure the ability of the control rod system to control reactivity.

Specification

(A1)

4.3 REACTIVITY CONTROL

Applicability

Applies to the surveillance requirements of the control rod system.

Objective

To verify the ability of the control rod system to control reactivity.

Specification

SDM 3.1.1

(A1)

~~A. Reactivity Limitations~~ SDM 3.1.1

3.1.1. ^{SHUTDOWN} ~~Reactivity margin - core loading~~ ^{SDM}

(A1)

~~A. Reactivity Limitations~~

~~1. Reactivity margin - core loading~~ (LAI)

LCO 3.1.1.a

(A2)

A sufficient number of control rods shall be operable so that the core could be made subcritical in the most reactive condition during the operating cycle with the strongest control rod fully withdrawn and all other operable control rods fully inserted.

SR 3.1.1.1.a

Proposed 2nd Frequency

Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to

(A3)

(A4)

demonstrate with a margin of 0.38% $\Delta k/k$ the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest operable control rod fully withdrawn and all other operable rods fully inserted.

(L1)

Proposed LCO 3.1.1.b

(M1)

Actions A, B, C, D + E

(M2)

SR 3.1.1.1 1st Frequency

Proposed SR 3.1.1.1.b

(L1)



APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.3.A.2 Reactivity margin - inoperable control rods (Cont'd)~~

~~4.3.A.2 Reactivity margin - inoperable control rods (Cont'd)~~

d. DELETED

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

d. The control rod accumulators shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

LCO 3.1.1

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a SHUTDOWN CONDITION within 24 hours.

See Justification for Changes for BFN ISTS 3.1.5

See Justification for Changes for BFN ISTS 3.1.3

Action B

12 MI

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



3.1
~~3.3/4.3~~ REACTIVITY CONTROL SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.3 REACTIVITY CONTROL

Applicability

Applies to the operational status of the control rod system.

Objective

To assure the ability of the control rod system to control reactivity.

Specification

A1

4.3 REACTIVITY CONTROL

Applicability

Applies to the surveillance requirements of the control rod system.

Objective

To verify the ability of the control rod system to control reactivity.

Specification

A1 ~~A. Reactivity Limitations~~ ^{SDM 3.1.1}
3.1.1 ~~Shutdown Reactivity Margin~~ ^(SDM) ~~Core Loading~~

A1 ~~A. Reactivity Limitations~~ ^{SDM 3.1.1}
~~1. Reactivity Margin~~ ~~Core Loading~~

LCO 3.1.1.a A sufficient number of control rods shall be OPERABLE so that the core could be made subcritical in the most reactive condition during the operating cycle with the strongest control rod fully withdrawn and all other OPERABLE control rods fully inserted.

SR 3.1.1.a Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.38% Δ k/k the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest OPERABLE control rod fully withdrawn and all other OPERABLE rods fully inserted.

A2

L1

Proposed LCO 3.1.1.b

Proposed 2nd frequency

A4

M1 ACTIONS A, B, C, D + E

M2 SR 3.1.1.1 1st frequency

Proposed SR 3.1.1.1.b

L1



~~3.3/A.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.3.A.2 Reactivity Margin - Inoperable Control Rods (Cont'd)~~

~~4.3.A.2 Reactivity Margin - Inoperable Control Rods (Cont'd)~~

d. DELETED

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

d. The control rod accumulators shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

LCO 2.1.1

1. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a SHUTDOWN CONDITION within 24 hours.

ACTION B

See Justification for Changes for BFN ISTS 3.1.5

See Justification for Changes for BFN ISTS 3.1.3

12

(M1)

UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



3.1

~~3.3/4.3 REACTIVITY CONTROL SYSTEMS~~

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.3 REACTIVITY CONTROL

Applicability

Applies to the operational status of the control rod system.

Objective

To assure the ability of the control rod system to control reactivity.

Specification

~~A. Reactivity Limitations~~ ^{SOM} 3.1.1

Shutdown (SDM)
3.1.1 1. ~~Reactivity margin~~ ~~core loading~~

(A1)

LCO 3.1.1.a

A sufficient number of control rods shall be OPERABLE so that the core could be made subcritical in the most reactive condition during the operating cycle with the strongest control rod fully withdrawn and all other OPERABLE control rods fully inserted.

(A2)

(L1)

Proposed LCO 3.1.1.b

(M1) ACTIONS A, B, C, D + E

4.3 REACTIVITY CONTROL

Applicability

Applies to the surveillance requirements of the control rod system.

Objective

To verify the ability of the control rod system to control reactivity.

Specification

~~A. Reactivity Limitations~~ ^{SOM} 3.1.1

(A1)

1. ~~Reactivity margin~~ ~~core loading~~

SR 3.1.1.1.a

Proposed 2nd Frequency

Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.38% Δ k/k the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest OPERABLE control rod fully withdrawn and all other OPERABLE rods fully inserted.

(LA1)

(A3)

(A4)

(M2) - SR 3.1.1.1 1st Frequency

Proposed SR 3.1.1.b

(L1)

APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.3.A.2 Reactivity margin - inoperable control rods (Cont'd)~~

~~4.3.A.2 Reactivity margin - inoperable control rods (Cont'd)~~

d. DELETED

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

d. The control rod accumulators shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

LCO 3.1.1

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a SHUTDOWN CONDITION within 24 hours.

ACTION B

12

m1

See Justification for change for BFN ISTS 3.1.5

See Justification for changes for BFN ISTS 3.1.3



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.1 - SHUTDOWN MARGIN (SDM)

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specification, NUREG 1433. These changes should make the BFN Technical Specifications easier for the operator (and other users) to read and understand. During the reformatting and renumbering process, no technical changes (either actual or interpretational) were made unless they were identified and justified.
- A2 The LCO has been reworded to include the actual limit. CTS describes how to demonstrate conformance to the limit, however the actual limit is located in the corresponding surveillance requirement. The required limit when the highest worth control rod is analytically determined (0.38% $\Delta k/k$) is included for clarity.
- A3 The proposed Surveillance Requirement provides a specific completion time to clarify when the SDM verification is to be completed. The intent of present Technical Specification 4.3.A.1 is to require the SDM test to be performed after in-vessel activities which could have altered SDM. More explicit wording is proposed to replace the activity referred to as "following a refueling outage when core alterations were performed." Most SDM tests are performed as an in-sequence critical. The proposed Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification. This interpretation is supported by the BWR Standard Technical Specifications, NUREG-1433. Since the proposed change clarifies the intent of the existing Surveillance Requirement, it is considered an administrative change.
- A4 Both limits described in Comment L1 below are also listed in the Surveillance. Thus, the limit for the highest worth rod determined by test (0.28% $\Delta k/k$) has been added.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 Currently, if SDM is not met the unit is placed in a Shutdown Condition (Mode 3) within 24 hours per CTS 3.3.A.2.f. Proposed Action B requires the plant to be placed in Mode 3 if SDM is not met. Proposed Actions C, D, and E for Modes 3, 4, and 5, are more restrictive than CTS since some additional action is required if SDM is not met (e.g., insert all insertable rods, suspend core alterations, initiate action to restore secondary containment to OPERABLE status, restore two standby gas



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.1 - SHUTDOWN MARGIN (SDM)

TECHNICAL CHANGE - MORE RESTRICTIVE (CONTINUED)

treatment subsystems to OPERABLE status and restore one isolation valve and associated instrumentation to OPERABLE status in each secondary containment penetration flow path with isolation valve(s) not isolated within 1 hour). The following changes were made to current Technical Specifications:

- If SDM is not met while the plant is in Mode 1 or 2, the proposed Actions (A and B) would require SDM to be restored in 6 hours or be in Mode 3 in the following 12 hours. Therefore, the proposed Specifications are more restrictive since only 18 hours is allowed to be in Mode 3. In addition, once in Mode 3, if the SDM was still not met, Action C would require the insertion of all insertable control rods. This action further enhances the available SDM. Since the plant was shut down to get to MODE 3, then the only action required is to insert all insertable control rods since secondary containment, standby gas treatment and isolation instrumentation are all required to be operable in MODE 3 anyway.
- If SDM is not met in MODE 4 or 5, new ACTIONS (ACTIONS D and E) are provided to initiate action to insert all insertable control rods (in core cells containing fuel), suspend CORE ALTERATIONS (if applicable), and to initiate actions within 1 hour to restore secondary containment, SGT System and the secondary containment isolation valves to OPERABLE status. The first two actions attempt to improve SDM, or at least to ensure SDM is not made worse, while the last three actions provide some protection from radioactive release if a SDM problem results in an inadvertent criticality.

These Actions are more restrictive since new requirements are added that currently do not exist.

- M2 An additional Surveillance Frequency for SDM verification (prior to each in-vessel fuel movement during fuel loading sequence) has been added to clarify the requirements necessary for assuring SDM during the refueling process. Because SDM is assumed in several refueling mode analyses in the FSAR, some measures must be taken to ensure the intermediate fuel loading patterns during refueling have adequate SDM. This change imposes a requirement where none is explicitly provided in the existing Technical Specifications. This new requirement does not, however,

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.1 - SHUTDOWN MARGIN (SDM)

TECHNICAL CHANGE - MORE RESTRICTIVE (CONTINUED)

require introducing tests or modes of operation of a new or different nature than currently exist.

As presented in the Bases corresponding to this requirement, this is best accomplished by analysis (rather than in-sequence criticals) because of the many changes in the core loading during a typical refueling. Bounding analyses may be used to demonstrate adequate SDM for the most reactive configurations during refueling thereby showing acceptability of the entire fuel movement sequence.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA1 Details of the methods to perform the Surveillance are relocated to the procedures. The requirement to verify the SDM is within the limit remains in the Surveillance. Procedures will be controlled by the licensee controlled programs.

"Specific"

L1 The current Technical Specifications indirectly require that the SDM be $\geq 0.38 \Delta k/k$ when the highest worth control rod is analytically determined. NUREG-1433 adds a less restrictive requirement that allows the SDM to be $\geq 0.28 \Delta k/k$ when the highest worth control rod is determined by test. This allows the SDM to be less when the highest worth control rod is determined by test. This is reasonable since the highest worth control rod is directly calculated. An actual measured value is obtained for the highest worth control rod versus an analytical one which may contain uncertainties that have to be accounted for in the analysis. The Surveillance Requirement also incorporates the new SDM value.

PAGE 3 OF 3



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

DEC 07 1994

~~3.3.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.3.E Reactivity Control~~

(A1)

ACTION
B

If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within 24 hours.

(M1)

24
12

3.3.F Scram Discharge Volume (SDV)

1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.
2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.
3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.

~~4.3.E Reactivity Control~~

(A1)

Surveillance requirements are as specified in 4.3.C and .D above.

4.3.F Scram Discharge Volume (SDV)

- 1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.
- 1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.
2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.
3. No additional surveillance required.

See Justification for Changes for BFN ISTS 3.1.8



NOV 03 1989

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.3.C. Scram Insertion Times

2. The average of the scram insertion times for the three fastest operable control rods of all groups of four control rods in a two-by-two array shall be no greater than:

<u>% Inserted From Fully Withdrawn</u>	<u>Avg. Scram Insertion Times (sec)</u>
5	0.398
20	0.954
50	2.120
90	3.800

3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

4.3.C. Scram Insertion Times

2. At 16-week intervals, 10% of the operable control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

see justification for changes to BFN ISTS 3.1.4

(M2)

OR control rod Replacement

(A1) ~~Reactivity Anomalies~~

LCO 3.1.2

(M3)

Modes 1+2

ACTIONS A+B

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk.

If this limit is exceeded, the reactor will be placed in the shutdown condition until the cause has been determined and corrective actions have been taken as appropriate.

(A2)

LCO 3.0.4

Proposed Required Action

A.1

(L1)

(A1)

~~Reactivity Anomalies~~

SR 3.1.2.1

(A3)

(A4)

During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions.

These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle.

(LA1)

At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon

(LA1)

appropriately corrected past data. This comparison will be made at least every full power month.

UNIT 2

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



NOV 03 1989

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.3.C. Scram Insertion Times

2. The average of the scram insertion times for the three fastest OPERABLE control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

3. The maximum scram insertion time for 90% insertion of any OPERABLE control rod shall not exceed 7.00 seconds.

4.3.C. Scram Insertion Times

2. At 16-week intervals, 10% of the OPERABLE control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

See Justification for Changes to BFN 1STS 3.1.4

(A1) ~~D. Reactivity Anomalies~~

LCO 3.1.2 - The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk.

If this limit is exceeded, the reactor will be placed in SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

M3 Modes 1+2

Actions A+B

SHUTDOWN CONDITION

Proposed Required Action A.1

A2 LCO 3.0.4

(A1) ~~D. Reactivity Anomalies~~
SR 3.1.2.1

During the STARTUP test program and STARTUP following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions.

These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle.

At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

(A3)

(A4)

or control rod replacement

(M2)

(LA1)

(LA1)

AMENDMENT NO. 175



DEC 07 1994

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.3.E. Reactivity Control~~

~~4.3.E. Reactivity Control~~

ACTION
B

If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within (M1) 12-24 hours.

(A1) Surveillance requirements are as specified in 4.3.C and .D above.

3.3.F. Scram Discharge Volume (SDV)

4.3.F. Scram Discharge Volume (SDV)

1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.
2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.
3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.

- 1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.
- 1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.
2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.
3. No additional surveillance required.

See Justification for Changes for BFN 1STS 3.1.8



UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



NOV 18 1988

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

3.3.C. Scram Insertion Times

- 2. The average of the scram insertion times for the three fastest OPERABLE control rods of all groups of four control rods in a two-by-two array shall be no greater than:

<u>% Inserted From Fully Withdrawn</u>	<u>Avg. Scram Insertion Times (sec)</u>
5	0.398
20	0.954
50	2.120
90	3.800

- 3. The maximum scram insertion time for 90% insertion of any OPERABLE control rod shall not exceed 7.00 seconds.

4.3.C. Scram Insertion Times

- 2. At 16-week intervals, 10% of the OPERABLE control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

See Justification for Changes to BFN 1STS 3.1.4

M2 OR Control Rod Replacement

(A1) ~~D. Reactivity Anomalies~~

LCD 3.1.2 → The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk.

(M3) Modes 1+2

Actions A+B { If this limit is exceeded, the reactor will be placed in the SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

LCD 3.0.4 →

(A2)

(L1) Proposed Required Action A.1

(A1) ~~D. Reactivity Anomalies~~ (A3) (A4)

SR3.1.2.1 During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

(LA1)

(LA1)



DEC 07 1994

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.3.E. Reactivity Control~~

(A1)

~~4.3.E. Reactivity Control~~

Action
B

If Specifications 3.3.C and 3.3.D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within 24 hours.

(ml) (12)

(A1)

Surveillance requirements are as specified in 4.3.C and 4.3.D above.

3.3.F. Scram Discharge Volume (SDV)

1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.

2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.

3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.

4.3.F. Scram Discharge Volume (SDV)

1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.

1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.

2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.

3. No additional surveillance required.

see justification for changes for BFN 1STS 3.1.B

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.2 - REACTIVITY ANOMALIES

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering is in accordance with the BWR/4 Standard Technical Specifications (STS), NUREG-1433. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the improved Technical Specifications, no technical changes (either actual or interpretational) to the TS were made unless they were identified and justified.
- A2 Proposed BFN ISTS LCO 3.0.4 does not permit entry into MODES unless the associated ACTIONS to be entered permit unlimited continued operation. The proposed Specification does not permit exit from MODE 3 (or entry into Mode 1 or 2) until the reactivity difference is restored. This is considered equivalent to the CTS wording of "until the cause has been determined and corrective actions have been taken as appropriate." Therefore, deleting these words are considered administrative.
- A3 Deleted "During the STARTUP test program" since this event has occurred and cannot occur again.
- A4 Proposed SR 3.1.2.1 provides a specific completion time for the reactivity anomaly surveillance to clarify when "during each startup" the test must be performed. The test is performed by comparing the actual rod configuration to the vendor provided predicted rod configuration as a function of cycle exposure while at steady state reactor power condition. A time frame of 24 hours after reaching these conditions is considered reasonable to allow performance of the required calculations for verification. This interpretation of the intent of the existing requirement is supported by the BWR Standard Technical Specification, NUREG-1433. Therefore, the proposed change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 CTS require the unit to be placed in the SHUTDOWN CONDITION (reactor in shutdown or refuel mode) if the specified limit is exceeded. CTS 3.3.E requires an orderly shutdown to be initiated and the reactor be placed in the SHUTDOWN CONDITION within 24 hours. Proposed BFN ISTS is more restrictive since it requires the unit to be placed in MODE 3 (Hot Shutdown) within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach MODE 3 from full power

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.2 - REACTIVITY ANOMALIES

TECHNICAL CHANGES - MORE RESTRICTIVE (CONTINUED)

conditions in an orderly manner and without challenging plant systems. Therefore, the proposed change is considered acceptable.

- M2 An additional requirement has been added to perform the Surveillance if control rods have been replaced, regardless of whether or not the unit is in a refueling outage. This ensures that any core change that could affect reactivity is evaluated properly.
- M3 The Applicability of the Reactivity Anomaly Specification has been expanded from during "power operation" to "MODES 1 and 2." This change represents an additional restriction on plant operations necessary to achieve consistency with safety analysis assumptions and NUREG-1433.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 Details of the methods to perform and purposes of the Surveillance are relocated to the Bases and procedures. The requirement, to verify the reactivity anomaly is within the limit, remains in the Surveillance. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee control programs.

"Specific"

- L1 Proposed Action A.1 provides a 72 hour time period to allow the core reactivity difference to be restored to within limits (i.e., to "perform an analysis to determine and explain the cause of the reactivity difference"). Typically, a reactivity anomaly would be indicative of incorrect analysis inputs or assumptions of fuel reactivity used in the analysis. A determination and explanation of the cause of the anomaly would normally involve an offsite fuel analysis and the fuel vendor. Contacting the vendor and obtaining the necessary input may require a time period much longer than one shift (particularly on weekends and holidays). Since shutdown margin has typically been demonstrated by test prior to reaching the conditions at which this surveillance is performed, the safety impact of the extended time for evaluation is negligible. Given these considerations, the BWR Standard Technical Specification, NUREG-1433 allows this time to be extended to 72 hours.



UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~
LCD 3.1.3 EACH CR SHALL BE OPERABLE

~~SURVEILLANCE REQUIREMENTS~~

~~3.3.A.2 Reactivity margin inoperable~~

4.3.A.2 Reactivity margin - inoperable control rods

control rods OPERABILITY

(MODS) 1+2

General Reorganization

ACTION A+B

Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to the GOLD SHUTDOWN CONDITION within 24 hours and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.2.c.

Action B
Action E

(L1)

Required ACTION A3

Within 72 hours

SR 3.1.3.2 a. Each partially or fully withdrawn OPERABLE control rod shall be exercised one notch at least once each week when operating above the power level cutoff of the RWM. In the event power operation is continuing with three or more inoperable control rods, this test shall be performed at least once each day, when operating above the power level cutoff of the RWM.

(M2)

(L2)

(HOT)

(M11)

(M3)

Required ACTION A.3

(L1)

(L4)

(M8)

Proposed Required Action A.1



APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

A1

~~3.3.A.2 Reactivity margin inoperable control rods (Control) OPERABILITY~~

4.3.A.2 Reactivity margin - inoperable control rods (Cont'd)

M4 → Add Required Action C.1

A1 - b. DELETED

Required Action C.2
 Required Action A.2
 Required Action B.1

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically. LA1

M1

A2

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

Required Action C.2

A3

c. When it is initially determined that a control rod is incapable of normal insertion a test shall be conducted to demonstrate that the cause of the malfunction is not a failure in the control rod drive mechanism. If this can be demonstrated an attempt to fully insert the control rod shall be made. If the control rod cannot be inserted and an investigation has demonstrated that the cause of failure is not a failed control rod drive mechanism collet housing, a shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined highest worth control rod capable of withdrawal fully withdrawn, and all other control rods capable of insertion fully inserted.



APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.3.A.2 ~~Reactivity margin - inoperable control rods (Cont'd)~~ OPERABILITY

4.3.A.2 Reactivity margin - inoperable control rods (Cont'd)

~~d. DELETED~~ (A1)

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable. (MS) Every 24 hours

d. The control rod accumulators shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

SR 3.1.3.1

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met, the reactor shall not be started, or if at power, the reactor shall be brought to a SHUTDOWN CONDITION within 24 hours.

Required Actions A.4, B.2 + C.1

(M10)

Condition D (L3)

See Justification for changes for Proposed Specification 3.1.5

Proposed Note to Condition D

(L5)

Condition E

(A4) LCO 3.0.4

(A5)

(M6)

(M7) Add 2nd Part of Proposed Condition E

Proposed Required Actions D.1 + D.2

(L3)

APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

A1

~~SURVEILLANCE REQUIREMENTS~~

~~3.3.B. Control Rods~~

1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the SHUTDOWN CONDITION when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

AB

ACTION C

LA1

A7

R1

2. The control rod drive housing support system shall be in place during REACTOR POWER OPERATION or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

~~4.3.B. Control Rods~~

~~SR 3.1.3.5~~

within 3 hours

within 4 hours

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

M4

a. Verify that the control rod is following the drive by observing any response in the nuclear instrumentation each time a rod is moved when the reactor is operating above the preset power level cutoff of the RWM.

LA2

~~SR 3.1.3.5~~

M9

AB

b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.



NOV 03 1989

~~3.3/4.3 REACTIVITY CONTROL~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.3.C. Scram Insertion Times

- 2. The average of the scram insertion times for the three fastest operable control rods of all groups of four control rods in a two-by-two array shall be no greater than:

<u>% Inserted From Fully Withdrawn</u>	<u>Avg. Scram Insertion Times (sec)</u>
5	0.398
20	0.954
50	2.120
90	3.800

(A2)

- 3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

SR 3.1.3.4

4.3.C. Scram Insertion Times

- 2. At 16-week intervals, 10% of the operable control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

See Justification for Changes for BFN ISTS 3.1.4

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be placed in the SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

See Justification for Changes for BFN ISTS 3.1.2

D. Reactivity Anomalies

During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.3.1.3 Each CR shall be OPERABLE

3.3.A.2 ~~Reactivity Margin - Inoperable Control Rods~~ OPERABILITY

4.3.A.2 ~~Reactivity Margin - Inoperable Control Rods~~

General Reorganization

ACTION A+B

Action B.2
Action E

L1

Required Action A.3

Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to the ~~SOLD~~ SHUTDOWN CONDITION within 24 hours and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.2.c.

within 72 hours

M8

Proposed Required Action A.1

SR 3.1.3.2
SR 3.1.3.5

A1

M2

L2

HOT

M11

M3

Required Action A.3

L1

a. Each partially or fully withdrawn OPERABLE control rod shall be exercised one notch at least once each week when operating above the power level cutoff of the RWM. In the event power operation is continuing with three or more inoperable control rods this test shall be performed at least once each day, when operating above the power level cutoff of the RWM.

L4



APR 30 1993

~~3.3/A.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

~~3.3.A.2 Reactivity Margin - Inoperable Control Rods (Cont'd)~~ OPERABILITY

4.3.A.2 Reactivity Margin - Inoperable Control Rods (Cont'd)

b. DELETED

M4

Add Required Action C.1

Required Action C.2
Required Action A.2
Required Action B.1

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically.

LAI

A2

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

Required Action C.2

c. When it is initially determined that a control rod is incapable of normal insertion a test shall be conducted to demonstrate that the cause of the malfunction is not a failure in the control rod drive mechanism. If this can be demonstrated an attempt to fully insert the control rod shall be made. If the control rod cannot be inserted and an investigation has demonstrated that the cause of failure is not a failed control rod drive mechanism collet housing, a shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined highest worth control rod capable of withdrawal fully withdrawn, and all other control rods capable of insertion fully inserted.

A5

APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.3.A.2 ~~Reactivity Margin - Inoperable Control Rods (Cont'd)~~ OPERABILITY

4.3.A.2 ~~Reactivity Margin - Inoperable Control Rods (Cont'd)~~

d. DELETED

e. Control rods with inoperable accumulators or those whose position cannot be positively determined, shall be considered inoperable.

SR3.1.3.1

M5 Every 24 hours

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started or if at power, the reactor shall be brought to a SHUTDOWN CONDITION within 24 hours.

Required Actions A1, B2 + C.1

M10

Condition D

L3

Proposed Note to Condition D L5

Condition E

A4 LCO 3.0.4

A5

M7 Add 2nd Part of Proposed Condition E

M6

Proposed Required Actions D.1 + D.2

L3

d. The control rod accumulators shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

See Justification for Changes for Proposed Specification 3.1.5



2.3/4.3 REACTIVITY CONTROL

APR 30 1993

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.3.B. Control Rods

4.3.B. Control Rods

SR 3.1.3.5

1. Each control rod shall be coupled to its drive or completely inserted, and the control rod directional control valves disarmed,

within 3 hours

within 4 hours

Action C

electrically. This requirement does not apply in the SHUTDOWN CONDITION when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

a. Verify that the control rod is following the drive by observing any response in the nuclear instrumentation each time a rod is moved when the reactor is operating above the preset power level cutoff of the RWM.

SR 3.1.3.5

b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance observe that the drive does not go to the overtravel position.

2. The control rod drive housing support system shall be in place during REACTOR POWER OPERATION or when the reactor coolant system is pressurized above atmospheric with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.



NOV 03 1989

(A1)

3.3.C. Scram Insertion Times

2. The average of the scram insertion times for the three fastest OPERABLE control rods of all groups of four control rods in a two-by-two array shall be no greater than:

<u>% Inserted From Fully Withdrawn</u>	<u>Avg. Scram Insertion Times (sec)</u>
5	0.398
20	0.954
50	2.120
90	3.800

(A2)

3. The maximum scram insertion time for 90% insertion of any OPERABLE control rod shall not exceed 7.00 seconds.

SR 3.1.3.4

4.3.C. Scram Insertion Times

2. At 16-week intervals, 10% of the OPERABLE control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

SEE JUSTIFICATION FOR CHANGES FOR BFN 1STS 3.1.4

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be placed in SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

D. Reactivity Anomalies

During the STARTUP test program and STARTUP following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

SEE JUSTIFICATION FOR CHANGES FOR BFN 1STS 3.1.2

AMENDMENT NO. 175



UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

LCD 3.1.3 Each CR shall be OPERABLE

(A1)

~~3.3.A.2 Reactivity margin inoperable control rods OPERABILITY~~ (Modes LS2)

4.3.A.2 ~~Reactivity margin in operable control rods~~

General reorganization

ACTION A.E.B

Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scan pressure the reactor shall be brought to the ~~COLD~~ SHUTDOWN CONDITION within 24 hours and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.2.c. within 72 hours

Action B.2
Action E

the reactor shall be brought to the ~~COLD~~ SHUTDOWN CONDITION within 24 hours and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.2.c. within 72 hours

SHUTDOWN CONDITION within 24 hours and shall not be started unless (1) investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.2.c. within 72 hours

investigation has demonstrated that the cause of the failure is not a failed control rod drive mechanism collet housing and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.2.c. within 72 hours

adequate shutdown margin has been demonstrated as required by Specification 4.3.A.2.c. within 72 hours

Required Action A.3

Required Actions A.2 and B.1

Required Action C.2

The control rod directional control valves for inoperable control rods shall be disarmed electrically

M8

Proposed Required Action A.1

M4

Add Required Action C.1

SR3.1.3.2 a. Each partially or fully withdrawn OPERABLE control rod shall be exercised one notch at least once each week when operating above the power level cutoff of the RWM. In the event power operation is continuing with three or more inoperable control rods, this test shall be performed at least once each day, when operating above the power level cutoff of the RWM.

M2

L2

(NOT)

M11

M3

Required Action A.3

L1

b. DELETED

L4



~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.3.A.2 ~~Reactivity margin - INOPERABLE control rods (Cont'd)~~

4.3.A.2 ~~Reactivity margin - INOPERABLE control rods (Cont'd)~~

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are INOPERABLE, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

A2

M4 { Required Action C.2

c. When it is initially determined that a control rod is incapable of normal insertion a test shall be conducted to demonstrate that the cause of the malfunction is not a failure in the control rod drive mechanism. If this can be demonstrated an attempt to fully insert the control rod shall be made. If the control rod cannot be inserted and an investigation has demonstrated that the cause of failure is not a failed control rod drive mechanism collet housing, a shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined highest worth control rod capable of withdrawal fully withdrawn, and all other control rods capable of insertion fully inserted.

A3



APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.3.A.2 Reactivity margin - inoperable control rods (Cont'd) OPERABILITY~~

4.3.A.2 Reactivity margin - inoperable control rods (Cont'd)

d. ~~DELETED~~ (A1)

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

SR 3.1.3.1

(M5) every 24 hours

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met, the reactor shall not be started, or if at power, the reactor shall be brought to a SHUTDOWN CONDITION within 24 hours.

Required Actions A.4, B.2 & C.1

(M10)

Condition D (L3)

Condition E

Proposed Note to Condition D (L5)

(A4) LCO 3.0.4

(A5)

(M1) Add 2nd part of Proposed Condition E (M6)

Proposed Required Actions D-1 & D.2 (L3)

d. The control rod accumulators shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

See Justification for Changes for proposed Specification 3.i-5

APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.3.B. Control Rods~~

~~4.3.B. Control Rods~~

SR 3-1.3.5

1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed ~~electrically~~.

A6

Action C

LA1

This requirement does not apply in the SHUTDOWN CONDITION when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

A7

RI

within 3 hours

within 4 hours

MA

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

a. Verify that the control rod is following the drive by observing any response in the nuclear instrumentation each time a rod is moved when the reactor is operating above the preset power level cutoff of the RWM.

LA2

SR 3.1.3.5 b.

When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

MA

A8

2. The control rod drive housing support system shall be in place during REACTOR POWER OPERATION or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.



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~~3.3/4.3 REACTIVITY CONTROL~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.3.C. Scram Insertion Times

2. The average of the scram insertion times for the three fastest OPERABLE control rods of all groups of four control rods in a two-by-two array shall be no greater than:

<u>% Inserted From Fully Withdrawn</u>	<u>Avg. Scram Insertion Times (sec)</u>
5	0.398
20	0.954
50	2.120
90	3.800

(A2)

SR 3.1.3.4

3. The maximum scram insertion time for 90% insertion of any OPERABLE control rod shall not exceed 7.00 seconds.

4.3.C. Scram Insertion Times

2. At 16-week intervals, 10% of the OPERABLE control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

See Justification for Changes For BFN ISTS 3.1.4

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be placed in the SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

See Justification for Changes for BFN ISTS 3.1-2

D. Reactivity Anomalies

During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY

ADMINISTRATIVE CHANGES

A1 All reformatting and renumbering is in accordance with the BWR/4 Standard Technical Specifications (STS), NUREG-1433. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the improved Technical Specifications, no technical changes (either actual or interpretational) to the TS were made unless they were identified and justified.

The organization of the Control Rod OPERABILITY specification is proposed to include all conditions that can affect the ability of the control rods to provide the necessary reactivity insertion and also to be simplified as follows:

- 1) a control rod is considered "inoperable" when it is degraded to the point that it cannot provide its scram function, when decoupled, or when its position is unknown. All inoperable control rods (except stuck rods) are required to be fully inserted and disarmed.
- 2) a control rod is considered "inoperable" and "stuck" if it is incapable of being inserted and requirements are retained to preserve shutdown margin for this situation.
- 3) a control rod is considered "slow" when it is capable of providing the scram function but may not be able to meet the assumed time limits.
- 4) and special considerations are provided for conformance to the banked position withdrawal sequence (BPWS) at less than 10% of rated thermal power.

The scram reactivity used in the safety analysis allows for a specified number of inoperable and slow scramming rods, and the control rod drop accident analysis provides additional considerations of the BPWS at low power levels.

Two "Notes" have been added. The first Note (at the start of the ACTIONS Table) provides more explicit instructions for proper application of the ACTIONS for Technical Specification compliance. The Note allows separate Condition entry for each control rod. In



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY

ADMINISTRATIVE CHANGES (CONTINUED)

conjunction with proposed Specification 1.3, "Completion Times," this Note provides direction consistent with the intent of existing Actions for inoperable control rods. The intent is to allow a specified period of time, for each inoperable control rod, to verify compliance with certain limits and, when necessary, fully insert and disarm.

The second Note, which is consistent with the requirements of proposed LCO 3.0.2, has been added to the ACTIONS and allows the RWM to be bypassed, if needed for continued operations, provided appropriate ACTIONS of proposed LCO 3.3.2.1 (RWM Specification) are taken. This is a human factors consideration to assure clarity of the requirement and allowance.

- A2 The requirement that control rods with scram times greater than those permitted by Specification 3.3.C.3 be considered inoperable (CTS 3.3.A.2.c) is included in proposed SR 3.1.3.4. The actions for control rods with scram times greater than the limit are more restrictive (see comment M4). Eliminating the separate Specification for excessive scram time by moving the requirement to another Specification, does not eliminate any requirements, or impose a new or different treatment of the requirements (other than those proposed in Comment M4). Therefore, this proposed change is considered administrative.
- A3 These requirements have been deleted since they are redundant to those currently found in BFN TS 3.3.A.2.a. Changes to that Specification are justified in the comments relating to that Specification. As such, this change is considered administrative.
- A4 This provision has been included in proposed BFN ISTS LCO 3.0.4 ("motherhood") and need not be repeated in individual Specifications. Proposed LCO 3.0.4 does not permit entry into a MODE or other specified condition in the Applicability except when the associated ACTIONS to be entered permit operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Therefore, removing this requirement is considered an administrative change.

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY**

ADMINISTRATIVE CHANGES (CONTINUED)

- A5 The "shutdown condition" has been more accurately described as "hot shutdown condition", i.e., MODE 3 in the proposed BFN ISTS. This is a human factors consideration to clarify the intent since currently "shutdown" could mean either hot or cold shutdown based on the definition provided in BFN TS 1.0.
- A6 The requirement that control rods be coupled to their drive mechanism is covered by proposed SR 3.1.3.5; thus, making it a requirement for control rods to be considered OPERABLE. Eliminating the current separate LCO for control rod coupling, by moving the surveillance and actions to proposed BFN ISTS 3.1.3, does not eliminate any requirements, or impose a new or different treatment of the requirements (other than those separately proposed). Therefore, this proposed change is considered administrative.
- A7 This requirement duplicates an identical and more appropriately placed requirement in existing Specification 3.10.A.6. Therefore, deletion of this requirement is considered administrative.
- A8 This Surveillance has been changed to more explicitly describe the requirement, which is to ensure that coupling is verified if maintenance on the control rod affected coupling. If maintenance is performed that does not affect coupling (e.g., HCU valve maintenance) there is no reason to perform testing.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 Proposed Required Actions A.2 and B.1 are comparable to CTS 3.3.A.2.b, which requires inoperable control rods (including stuck control rods) to be disarmed. Two hours is allowed to disarm withdrawn control rods that are stuck. Since CTSs do not provide a maximum time limit, the proposed change is considered more restrictive.
- M2 Proposed SR 3.1.3.2 and SR 3.1.3.3 require control rods to be inserted rather than the existing requirement of exercised, which could be met by control rod withdrawal. It is conceivable that a mechanism causing binding of the control rod that prevents insertion could exist such that a withdrawal test would not detect the problem. Since the purpose of the test is to assure scram insertion capability, restricting the test



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - MORE RESTRICTIVE (CONTINUED)

- to only allow control rod insertion provides an increased likelihood of this test detecting a problem that impacts this capability.
- M3 This Surveillance has been moved to Required Action A.3. In addition, this is now required when as few as one control rod is immovable.
- M4 Added Required Action C.1, which requires an inoperable rod (unless stuck) to be fully inserted within 3 hours and disarmed within 4 hours. Placed a time limit on existing TS 3.3.A.2.b for disarming control rods (Required Action C.2) and existing TS 3.3.B.1 for inserting and disarming control rods. This is more restrictive than current requirements, which allow the rod to remain withdrawn when inoperable. Also, this is more restrictive since the ISTS requires disarming even if rod can be inserted with drive pressure. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed to prevent inadvertent withdrawal during subsequent operation. Reference related Comment A1. Since existing Technical Specifications do not provide a maximum time limit, the proposed change is considered more restrictive.
- M5 This requirement has been modified to require the position of each control rod to be verified every 24 hours (proposed SR 3.1.3.1). Current requirements do not have a specific Surveillance for this requirement.
- M6 Proposed Required Actions D.1 and D.2 allow 4 hours to restore compliance with the Specification (i.e., restore control rods to operable status or restore compliance with the BPWS). This change is considered more restrictive since the current time to reach a shutdown condition (MODE 3) has been reduced from 24 hours to 12 hours (per proposed Required Action E.1). Since the total time to reach a shutdown condition has been effectively changed from 24 hours to 16 hours (4 to restore and 12 to reach MODE 3), this proposed change is considered more restrictive.
- M7 A new Condition has been added (second part of proposed Condition E) requiring an shutdown (i.e., be in MODE 3 within 12 hours) if 9 or more control rods are inoperable. Currently, 9 control rods can be inoperable, provided they are separated by four operable control rods, without requiring shutdown.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - MORE RESTRICTIVE (CONTINUED)

- M8 Proposed Required Action A.1 has been added to confirm that when a control rod is found stuck, it is properly separated from "slow" control rods. The other Required Actions of ACTION A were renumbered to reflect the insertion of A.1.

The scram reactivity analysis assumes, among other things that there are two "slow" control rods adjacent to one another, a third control rod is stuck in the withdrawn position, and a fourth control rod fails to scram during the transient/accident analysis (the single failure). However, the analysis does not assume that the original stuck control rod is adjacent to the two "slow" rods or to another "slow" control rod. If this occurs, the local scram reactivity rate assumed in the analysis might not be met.

- M9 Changed Frequency for verifying coupling to each time the rod is withdrawn to the full out position, not just the first time after each refueling outage.

- M10 Existing Specification 3.3.A.2.f requires that inoperable (and stuck) control rods be positioned such that SDM requirements (3.3.A.1) are maintained. Proposed Required Actions A.4, B.2 and C.1 for LCO 3.1.3 requires that with one stuck rod (A.4) that shutdown margin be verified within 72 hours (Justification L1), with more than one stuck rod (B.2) that the reactor be in Hot Shutdown within 12 hours, and with one or more inoperable rods (C.1) that each inoperable rod be fully inserted. By allowing one stuck rod and by requiring that all insertable control rods be fully inserted, the proposed Required Actions provide greater assurance that SDM is maintained than the requirement for verifying SDM for multiple rods withdrawn.

- M11 The current time to reach a non-applicable condition has been reduced from 24 hours to reach Cold Shutdown (MODE 4) to 12 hours to reach MODE 3 (per Required Action E.1). This change is more restrictive because all rods must be fully inserted in 12 hours instead of the currently required 24 hours. Cooling the unit down (proceeding from MODE 3 to MODE 4) does not provide any additional margin and, in some cases, could be counter productive since positive reactivity is inserted during cooldown.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 Details of the methods of disarming control rod drives (CRDs) are relocated to the Bases and procedures. The requirement to disarm the CRD remains in the Specification.
- LA2 Details of the methods of verifying control rod coupling are relocated to plant procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 Proposed Action A allows continued operation with one withdrawn control rod stuck provided that Shutdown Margin is demonstrated. With a single control rod stuck in a withdrawn position, the remaining control rods are capable of providing the required scram and shutdown reactivity. Failure to reach COLD SHUTDOWN is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram. Even with the postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach and maintain HOT SHUTDOWN conditions. Required Action A.3 of LCO 3.1.3 performs a notch test on each remaining control rod to ensure that no additional control rods are stuck. The reason for the failure (e.g., failed collet housing) is not significant provided all other rods are tested to ensure a like failure has not occurred. Given these considerations, the 72 hours allowed to demonstrate SHUTDOWN MARGIN is considered reasonable to perform the analysis or test.
- L2 Proposed SR 3.1.3.3 extends the surveillance that verifies control rods are not stuck from 7 days to 31 days for control rods that are not fully withdrawn. This is consistent with the BWR Standard Technical Specifications, NUREG-1433. Partially withdrawn control rods have a significantly greater effect on core flux distribution than do fully withdrawn control rods. Historically, power reductions are required each week to perform the test on partially withdrawn control rods. The impact of testing on plant capacity is deemed excessive given the following considerations:

PAGE 6 OF 8



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - LESS RESTRICTIVE (CONTINUED)

- 1) At full power a large percentage of control rods (typically 80 - 90%) are fully withdrawn and would continue to be exercised each week. This represents a significant sample size when looking for an unexpected random event.
 - 2) Operating experience has shown that "stuck" control rods are an extremely rare event while operating.
 - 3) Should a stuck rod be discovered, 100% of the remaining control rods (even partially withdrawn) must be tested within 24 hours (proposed Required Action A.3).
- L3 The requirement that no more than one control rod in any 5 x 5 array may be inoperable (at least four operable control rods must separate any two inoperable ones) is proposed to be changed to allow inoperable control rods to be separated by two operable control rods. This is consistent with the safety analyses associated with this limitation. Proposed ACTION D addresses the condition when the reactor is $\leq 10\%$ RTP and two or more inoperable control rods are not in compliance with the BPWS and not separated by two or more operable control rods. The required action is to restore compliance with the BPWS within 4 hours or restore the control rod to operable status within 4 hours. Inoperable control rod separation requirements are required at $\leq 10\%$ RTP because of Control Rod Drop Accident (CRDA) concerns related to control rod worth. Above 10% RTP, control rod worths that are of concern for the CRDA are not possible. The proposed two operable control rod separation criteria in ACTION D is acceptable for the BPWS analysis and therefore, is acceptable for use in the proposed TS.
- L4 The current TSs require a daily notch test in the event power operation is continuing with three or more inoperable control rods and the plant is operating at $> 30\%$ RTP. The proposed TS only require the control rod notch test in the case of a single stuck control rod, and only once within 24 hours. The purpose of the control rod notch test on each withdrawn operable control rod is to ensure that a generic problem does not exist and that control rod insertion capability remains. The single performance of the control rod notch test satisfies the same function as the daily notch test of the current TS without requiring the additional testing.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - LESS RESTRICTIVE (CONTINUED)

- L5 The requirement (control rod separation requirement) associated with the proposed Note to Condition D (which limits the requirement to $\leq 10\%$ RTP) is necessary to ensure the rod pattern is in compliance with BPWS. This ensures that a rod drop accident will not result in excessive local power in a fuel bundle. Analysis has shown that inoperable control rod distribution is not a problem when $> 10\%$ RTP. The analysis is described in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Revision 8, Amendment 17. This analysis also showed that the inoperable control rod distribution is needed at $\leq 1\%$ RTP, which is broader than the current requirement for reactor power operation. The inoperable control rod distribution requirement has been modified to include this new restriction. Therefore, any decrease in safety by eliminating the distribution requirement $> 10\%$ RTP, is offset by the added safety of requiring inoperable control rod distribution at lower power when a rod drop accident can impact fuel design limits.

RELOCATED SPECIFICATIONS

- R1 CRD OPERABILITY requirements (CTS 3.3.B.2) currently include requirements for the CRD housing support to be in place. These requirements have been relocated to plant procedures. The CRD Housing Support does support CRD operability which is part of the primary success path. Having the CRD Housing Support out of place does impact CRD operability. It is indirectly covered in ISTS 3.2.3 in the blanket action for a control rod being inoperable for any other reason. There is no need to duplicate requirements in a subsystem LCO. Relocation of this LCO is appropriate since plant configuration (the control rod housing support in place) would be control by post maintenance procedures.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
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APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1) 3.3.C. Scram Insertion Times

LCO 3.1.4

Table 3.1.4-1

Note (A) of TS 3.1.4-1

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all OPERABLE control rods in the reactor power operation condition shall be no greater than:

except where noted

(M1)

(M2)

% Inserted From Fully Withdrawn Avg. Scram Insertion Times (sec)

5	0.375
20	0.90
50	2.0
90	3.500

(A2)

(M1)

Add Table 3.1.4-1, Notes 1

(A1) 4.3.C. Scram Insertion Times

SR 3.1.4.1

(A3)

1. After each refueling outage, all OPERABLE rods shall be scram-time tested from the fully withdrawn position with the nuclear system pressure above 800 psig. This testing shall be completed prior to exceeding 40% power.

(A5)

(Z)

Below 10% power, only rods in those sequences which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram-time tested.

(LA1)

2nd Frequency of Proposed SR 3.1.4.1

(M3)

NOV 03 1989

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.3.G. Scram Insertion Times~~

A1

M1

2. The average of the scram insertion times for the three fastest operable control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

A4

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be placed in the SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

See Justification for Changes for Proposed BFN ISTS 3.1.2

~~4.3.G. Scram Insertion Times~~
SR 3.1.4.2

LA2

2. (L1) At 16-week intervals, 10% of the operable control rod drives shall be scram-timed above 800 psig.

Z

Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

A5

LA3

M6

SR 3.1.4.3

M4

SR 3.1.4.4

D. Reactivity Anomalies

During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

DEC 07 1994

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

41 3.3.E ~~Reactivity Control~~

4.3.E ~~Reactivity Control~~

If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within 24¹² hours.

AI Surveillance requirements are as specified in 4.3.C and .D above.

ACTION A

MS

3.3.F Scram Discharge Volume (SDV)

4.3.F Scram Discharge Volume (SDV)

1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.

1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.

2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.

1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.

3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.

2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.

3. No additional surveillance required.

See justification for changes for BFN ISTS 3.1.8

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



APR 30 1993

3.3/4.3 REACTIVITY CONTROL

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

A1 3.3.8. Scram Insertion Times

A1 4.3.6. Scram Insertion Times
SR 3.1.4.1

LCO 3.1.4

Table 3.1.4-1

Note (A) of T3.1.4-1

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all OPERABLE control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.500

Exempt where noted

M1

M2

A2

Adel Table 3.1.4-1, Notes. 1

M1

M1

Proposed Note for SRs

1. After each refueling outage, all OPERABLE rods shall be scram-time tested from the fully withdrawn position with the nuclear system pressure above 800 psig. This testing shall be completed prior to exceeding 40% power.

LAI

Below 10% power, only rods in those sequences which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram-time tested.

2nd Frequency of proposed SR 3.1.4.1

M3

A3

AS

NOV 03 1989

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.3.C. Scram Insertion Times~~ (A1)

MI 2. The average of the scram insertion times for the three fastest OPERABLE control rods of all groups of four control rods in a two-by-two array shall be no greater than:

<u>% Inserted From Fully Withdrawn</u>	<u>Avg. Scram Insertion Times (sec)</u>
5	0.398
20	0.954
50	2.120
90	3.800

AY 3. The maximum scram insertion time for 90% insertion of any OPERABLE control rod shall not exceed 7.00 seconds.

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be placed in SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

SEE JUSTIFICATION FOR CHANGES FOR PROPOSED BFN ISTS 3.1.2

~~4.3.C. Scram Insertion Times~~

SR 3.1.4.2

2. At 16-week intervals 10% of the OPERABLE control rod drives shall be scram-timed above 800 psig. (LA2) (AS)

LA3 Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

MG SR 3.1.4.3

M4 SR 3.1.4.4

D. Reactivity Anomalies

During the STARTUP test program and STARTUP following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

AMENDMENT NO. 175

DEC 07 1994

~~2.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

(A1)

~~3.3.E. Reactivity Control~~

ACTION A

If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within ~~24~~ 12 hours.

(MS)

3.3.F. Scram Discharge Volume (SDV)

1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.
2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.
3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.

See Justification for Changes for BFN 15TS 3.1.8

4.3.E. Reactivity Control

(A1)

Surveillance requirements are as specified in 4.3.C and .D above.

4.3.F. Scram Discharge Volume (SDV)

- 1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.
- 1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.
2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.
3. No additional surveillance required.



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1) 3.3.6. Scram Insertion Times

LCO 3.1.4

Table 3.1.4-1

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all OPERABLE control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.5

Except where noted

M1

M2

A2

(M1) Add Table 3.1.4-1, Notes 1

(A1)

4.3.C. Scram Insertion Times SR 3.1.4.1

(A3)

1. After each refueling outage, all OPERABLE rods shall be scram-time tested from the fully withdrawn position with the nuclear system pressure above 800 psig. This testing shall be completed prior to exceeding 40% power. Below 10% power, only rods in those sequences which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram-time tested.

M1 Proposed note for SRS

A5

≥

(LA1) Below 10% power, only rods in those sequences which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram-time tested.

2nd Frequency of Proposed SR 3.1.4.1

M3

NOV 18 1988

LA2

~~3.3/A.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.3.C. Scram Insertion Times~~

~~A.3.C. Scram Insertion Times~~
SR 3.1.4.2

(M1)

2. The average of the scram insertion times for the three fastest OPERABLE control rods of all groups of four control rods in a two-by-two array shall be no greater than:

<u>% Inserted From Fully Withdrawn</u>	<u>Avg. Scram Insertion Times (sec)</u>
5	0.398
20	0.954
50	2.120
90	3.800

2. At 16-week intervals, 10% of the OPERABLE control rod drives shall be scram-timed above 800 psig. (L1)

Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained. (LA3, M6)

SR 3.1.4.3

SR 3.1.4.4 (M4)

(A4) 3. The maximum scram insertion time for 90% insertion of any OPERABLE control rod shall not exceed 7.00 seconds.

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be placed in the SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

D. Reactivity Anomalies

During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month. (A5)

See Justification for changes for Proposed BFN ISTS 3.1-2



~~3.3/4.3 REACTIVITY CONTROL~~

DEC 07 1994

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

(A1) ~~3.3.E. Reactivity Control~~

4.3.E. Reactivity Control

ACTION A

MS

If Specifications 3.3.C and 3.3.D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN (HOT) CONDITION within ~~24~~^{2 1/2} hours.

(A1) Surveillance requirements are as specified in 4.3.C and 4.3.D above.

3.3.F. Scram Discharge Volume (SDV)

4.3.F. Scram Discharge Volume (SDV)

1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.
2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.
3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.

- 1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.
- 1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.
2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.
3. No additional surveillance required.

see justification for changes for BFN 15 TS 3.1.8



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.4 - CONTROL ROD SCRAM TIMES

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering is in accordance with the BWR/4 Standard Technical Specifications (STS), NUREG-1433. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the improved Technical Specifications, no technical changes (either actual or interpretational) to the TS were made unless they were identified and justified.
- A2 CTS lists the position of the control rod in terms of % inserted from the fully withdrawn position. Proposed BFN ISTS Table 3.1.4-1 list the position in terms of notch position. These positions are equivalent (to the next nearest measured notch position) except expressed in different terms.
- A3 The Surveillance Frequency has been modified to require testing after fuel movement within the reactor pressure vessel. This is equivalent to after each refueling outage, which implies that fuel has been moved.
- A4 Proposed Specification 3.1.4 retains the current maximum scram time requirement (7 seconds) of Specification 3.3.C.3 for the purpose of defining the threshold between a "slow" control rod and an inoperable control rod. Note 2 of proposed Table 3.1.4-1 ensures that a control rod is not inadvertently considered "slow" when scram time exceeds 7 seconds. Proposed SR 3.1.3.4 verifies each control rod's maximum scram time is ≤ 7 seconds or it is considered inoperable.
- A5 CTS 4.3.C.1 & 2 requires scram time testing to be performed at > 800 psig. SRs 3.1.4.1 & 2 require testing to be performed at ≥ 800 psig. The requirement to perform this testing at pressure = 800 psig is slightly less restrictive since the SRs can be performed over a slightly broader pressure range. However, since the change is so minor it has been categorized as administrative. The proposed change is consistent with BWR/4 Standard Technical Specifications (NUREG-1433).

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 The LCO for Control Rod Scram times ensures that the negative scram reactivity assumed in the DBA and transient analysis is met. Current BFN Unit 2 Technical Specifications accomplish this by specifying the



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.4 - CONTROL ROD SCRAM TIMES

TECHNICAL CHANGES - MORE RESTRICTIVE (CONTINUED)

maximum individual scram times (7.0 seconds), average scram times and local scram times (four control rod group).

The design basis transient analysis assumes all control rods scram at the same speed. If all control rods scram at least as fast as the analytical limit, the scram reactivity assumed in the DBA and transient analysis is met. A distribution of scram times (some slower and some faster than the analytical limit) can also provide adequate scram reactivity. The more control rods that scram slower than the analytical limit, the faster the remaining control rods must scram to compensate for the reduced reactivity of the slower control rods. Proposed BFN ISTS 3.1.4 incorporates this principle to ensure adequate scram reactivity by specifying scram time limits for individual control rods instead of limits on average or four control rod groups. This methodology is similar to that being used for the BWR/6 STS. The LCO scram time limits have margin to the analytical scram time limits to allow for a specified number and distribution of slow control rods, a single stuck control rod and an assumed single failure.

The proposed LCO specifies the number and distribution of "slow" control rods allowed that will still ensure the analytical scram reactivity assumptions are satisfied. If the number of "slow" rods is excessive (>13) or do not meet the distribution requirements, the unit must be shutdown. This change is more restrictive since the proposed individual times are more restrictive than the average times. Currently, the "average" time of all rods or a group can be improved by a few fast scrambling rods, even when there may be more than 13 "slow" rods. The proposed specification limits the number of slow rods to 13 and ensures each slow rod is separated by two operable rods.

Table 3.1.4-1 is modified by a note (Note A), which state that control rods with scram times not within limits of the table are considered slow and those with times greater than 7 seconds are considered inoperable as required by SR 3.1.3.4.

In addition, a note has been added to the Surveillance Requirements Table requiring that, during a single control rod scram time Surveillance, the CRD pumps be isolated from the associated accumulator. This ensures that accumulator pressure alone is scrambling the rod, not the CRD pump pressure (which can improve the scram times).



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.4 - CONTROL ROD SCRAM TIMES

TECHNICAL CHANGES - MORE RESTRICTIVE (CONTINUED)

- M2 Proposed BFN ISTS 3.1.4 applicability of MODES 1 and 2 includes power levels \leq 1% RTP when first pulling rods to go critical. The applicability for current TS 3.3.C.1 of "in the reactor power operation condition" is defined by CTS Definition 1.0.H as any operation in the STARTUP/HOT STANDBY or RUN MODE with the reactor critical and above 1 percent rated power. Therefore, the proposed applicability is more restrictive.
- M3 Added a Frequency for performing scram time tests on all control rods prior to exceeding 40% RTP. This Frequency requires these tests after each reactor shutdown \geq 120 days regardless of whether refueling occurred.
- M4 Added Surveillance Requirement (SR 3.1.4.4) that requires a scram time test after work on a control rod or CRD that could affect the scram time. The Surveillance requires a scram time test after reactor pressure has reached \geq 800 psig and prior to exceeding 40% RTP.
- M5 CTS require the unit to be placed in the SHUTDOWN CONDITION (reactor in shutdown or refuel mode) if the specified limit is exceeded. CTS 3.3.E requires an orderly shutdown to be initiated and the reactor be placed in the SHUTDOWN CONDITION within 24 hours. Proposed BFN ISTS is more restrictive since it requires the unit to be placed in MODE 3 (Hot Shutdown) within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. Therefore, the proposed change is considered acceptable.
- M6 Added Surveillance Requirement (SR 3.1.4.3) that requires a scram test after work on a control rod or CRD that could affect the scram time prior to declaring the control rod OPERABLE with reactor steam dome pressure $<$ 800 psig. This scram test is performed with the rod inserted and the accumulator drained and isolated. The head of the water from the reactor acting on the under piston area will cause the rod to insert to overtravel and demonstrate that the scram valves open and the scram discharge volume exhaust path is open. This shows that the rod can be scrambled without subjecting it to unnecessary stress.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.4 - CONTROL ROD SCRAM TIMES

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 CTSs allow only rods in those sequences which were fully withdrawn in the region from 100% rod density to 50% rod density to be scram-time tested when below 10% power. This ensures that in-sequence fully withdrawn control rods are tested at low power where most rod worth is a concern. The Rod Pattern Control Specification and RWM ensure proper CR sequences are followed. Details of the restrictions, methods and purpose of the Surveillance are relocated to plant procedures. The requirement to perform scram time testing remains in the surveillance.
- LA2 Proposed SR 3.1.4.2 requires a "representative sample" of control rods to be tested each 120 days of operation instead of the currently required 10% of the OPERABLE control rods (CTS SR 4.3.C.2). The proposed change adopts the position of the BWR Standard Technical Specifications, NUREG 1433, that these details be located in plant procedures and summarized in the Bases for the Surveillance.
- LA3 Details of the method to perform or the purpose of the Surveillance are relocated to plant procedures. The requirement to perform scram time testing remains in the surveillance. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 Proposed SR 3.1.4.2 is performed at 120 days cumulative operation in MODE 1 versus the CTS requirement of 16-week intervals. Since the proposed frequency is longer than 16-weeks it is considered less restrictive. The 120 day Frequency is based on operating experience that has shown control rod scram times do not significantly change over an operating cycle. This Frequency is reasonable based on the additional Surveillances done on CRDs at more frequent intervals in accordance with LCO 3.1.3 and LCO 3.1.5, "Control Rod Scram Accumulators."

PAGE 4 OF 4

UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1) 3.3.A.2 ~~Reactivity margin inoperable control rods (Cont'd)~~

4.3.A.2 ~~Reactivity margin in operable control rods (Cont'd)~~

LCO 3.1.5

d. DELETED

(See Justification for changes for BFN 1STS 3.1.3)

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

(L1)

Proposed ACTIONS A, B, C+D

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a SHUTDOWN CONDITION within 24 hours.

SR 3.1.5.1

(Verify each)
d. The control rod accumulator shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

Pressure is \geq 940 psig

(LAI)

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



APR 30 1993

3.3/4.3 REACTIVITY CONTROL

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

(AI) ~~3.3.A.2 Reactivity Margin Inoperable Control Rods (Cont'd)~~

(AI) ~~4.3.A.2 Reactivity Margin Inoperable Control Rods (Cont'd)~~

LCO 3.1.5

(See Justification for changes for BFN ISTS 3.1.3)

SR 3.1.5.1

d. DELETED
e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

d. ^{Verify each} The control rod accumulator shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

Proposed ACTIONS A, B, C + D

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a SHUTDOWN CONDITION within 24 hours.

pressure is \geq 940 psig

(LAI)



UNIT 3
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



APR 30 1993

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

~~3.3.A.2 Reactivity margin - inoperable control rods (Cont'd)~~

(A1)

~~4.3.A.2 Reactivity margin - inoperable control rods (Cont'd)~~

LCO 3.1.5

d. DELETED

SR3-1.5.1.d. ^{Verify each} ~~The control rod accumulator shall be determined OPERABLE at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.~~

(L1)

Proposed Actions A, B, C, D

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

(A2)

Pressure is \geq 940 PSI g

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5x5 array may be inoperable (at least 4 OPERABLE control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a SHUTDOWN CONDITION within 24 hours.

(LA1)

See Justification for changes for BFN 1STS 3.1.3



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering is in accordance with the BWR/4 Standard Technical Specifications (STS), NUREG-1433. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the improved Technical Specifications, no technical changes (either actual or interpretational) to the TS were made unless they were identified and justified.
- A2 Proposed SR 3.1.5.1 requires that the accumulator pressure be checked to ensure adequate accumulator pressure exists to provide sufficient scram force. This satisfies the intent of the existing surveillance. Therefore, the proposed changes are considered administrative.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 Details of the method to perform or the purpose of the Surveillance are relocated to plant procedures. The requirement to ensure adequate scram pressure exists, to provide the necessary scram force, remains in the surveillance. The primary safety concern is accumulator pressure. Increasing water level indicates deterioration of the accumulator piston seal to the nitrogen side. The requirement for verification that the level detectors are not in alarm has been relocated to plant procedures. Changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 Proposed BFN ISTS 3.1.5, which replaces BFN TS 3.3.A.2.e, allows a short out of service time for the accumulators (Actions A and B also allow the control rods to be declared "slow" instead of inoperable) prior to declaring the associated control rods inoperable provided that proposed ACTIONS A, B, C and D are met. The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each control rod scram accumulator. This is acceptable since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable accumulator. Complying with the Required Actions may allow for continued operation and subsequent inoperable accumulators governed by subsequent Condition entry and application of associated Required Actions.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

TECHNICAL CHANGES - LESS RESTRICTIVE (CONTINUED)

Proposed Action A allows one control rod scram accumulator to be inoperable for up to eight hours when reactor steam dome pressure is ≥ 900 psig before declaring the associated control rod scram time slow or declaring the associated control rod inoperable. With one accumulator inoperable, the control rod may be declared "slow," since the control rod will still scram at the reactor operating pressure but may not satisfy the required scram times. Since the existing action (BFN TS 3.3.A.2.c) to declare the control rod inoperable would allow the control rod to remain withdrawn and not disarmed, the proposed action to declare the control rod "slow" is essentially equivalent. The proposed limits and allowance for numbers and distribution of inoperable and "slow" control rods (found in proposed LCOs 3.1.3 and 3.1.4 respectively) are appropriately applied to control rods with inoperable accumulators whether declared inoperable or "slow." Required Action A.1 is modified by a Note indicating that declaring the control rod "slow" only applies if the associated control scram time was within the limits during the last test.

Proposed Action B allows two or more control rod scram accumulators to be inoperable for one hour when reactor steam dome pressure is ≥ 900 psig provided charging pressure is restored within 20 minutes. Condition B requires that Required Action B.1 be taken in conjunction with Required Action B.2.1 or B.2.2. Required Action B.1 addresses the situation where additional accumulators may be rapidly becoming inoperable due to loss of charging pressure (charging pressure must be restored within 20 minutes). Required Actions B.2.1 and B.2.2 require that the associated control rods be declared "slow" or inoperable within one hour, which provides a reasonable time to attempt investigation and restoration of the inoperable accumulator. Since reactor pressure is adequate to assure the scram function and charging pressure is adequate, the proposed 1 hour extension is not significant.

Proposed Action C allows one or more accumulators to be inoperable with reactor steam dome pressure ≤ 900 psig provided that Required Action C.1 (verify that all control rods associated with inoperable accumulators are fully inserted) is taken immediately upon discovery of charging water header pressure < 940 psig and Required Action C.2 (declare the associated control rod inoperable) is taken within one hour. Required Action C.1 must be completed immediately since adequate scram pressure is not guaranteed (i.e., reactor steam dome pressure ≤ 900 psig).

JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

TECHNICAL CHANGES - LESS RESTRICTIVE (CONTINUED)

Once verification of adequate charging pressure is made (20 minutes is provided) and considering reactor pressure is adequate to assure the scram function of the control rods with inoperable accumulators, the proposed 1 hour completion time is not significant. In additions, since the reactor pressure may not be adequate to scram the rods in the proper time, Action C does not allow the rods to be declared "slow" (as allowed by Actions A and B).

Proposed Action D requires an immediate scram if any Required Action or associated Completion time can not be met. This ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. This Required Action is modified by a Note stating that the action is not applicable if all control rods associated with the inoperable scram accumulators are fully inserted, since the function of the control rods has been performed.





INSERT PROPOSED NEW SPECIFICATION 3.1.6

Insert new Specification 3.1.6, "Rod Pattern Control," as shown in the BFN Unit 2 Improved Technical Specifications.

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.6 - ROD PATTERN CONTROL**

TECHNICAL CHANGES - LESS RESTRICTIVE

- L1 A specific requirement for control rods to be in compliance with the BPWS during operation at low power is proposed as TS 3.1.6. This proposed specification also contains an allowance (Actions to LCO 3.1.6) for a limited number of out-of-sequence Operable control rods, which is presented in the BWR Standard Technical Specification, NUREG-1433, and also proposed to be included in the revised Technical Specifications. The Actions allow up to 8 out-of-sequence operable control rods (separate from any inoperable out-of-sequence control rods) to be returned to their correct position within 8 hours. This allowance for correction is proposed in recognition of the occurrence of such events as "double-notch" rod withdrawals, and minor misalignment of rod pattern during CRD hydraulic transients (control rod drift due to excessive cooling water pressure) or during a plant shutdown. These events can introduce out-of-sequence control rod patterns which the RWM was unable to preclude, even though the RWM was functioning as designed.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



~~3.4.4 STANDBY LIQUID CONTROL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the operating status of the Standby Liquid Control System.

Objective

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

Specification

A. ~~Normal System Availability~~

- 1. Except as specified in 3.4.3.1, the Standby Liquid Control System shall be OPERABLE at all times when there is fuel in the reactor vessel and the reactor is not in a shutdown condition with Specification 3.3.A.1 satisfied.

LCO 3.1.7

(L1)

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective

To verify the operability of the Standby Liquid Control System.

Specification

A. ~~Normal System Availability~~

The operability of the Standby Liquid Control System shall be verified by the performance of the following tests:

- 1. Verify pump OPERABILITY in accordance with Specification 1.0.MM.
- 2. At least once during each operating cycle:

(A2)

a. Check that the setting of the system relief valves is 1,425 ± 75 psig.

(LA1)

SR 3.1.7.8

b. Manually initiate the system, except explosive valves. Visually verify flow by pumping boron solution through the recirculation path and back to the Standby Liquid Control Solution Tank. After pumping boron solution, the system shall be flushed with demineralized water. Verify minimum

(LA2)

SR 3.1.7.6



(A1) ~~4.4.4 Normal System Applicability~~ e 1

~~4.4.4.2.b. (Cont'd)~~ e (A1)

SR 3.1.7.6 pump flow rate of 39 gpm against a system head of 1275 psig/ft pumping demineralized water from the Standby Liquid Control Test Tank

LA2

SR 3.1.7.7 c. ~~Manually~~ initiate one of the Standby Liquid Control System loops and pump ~~demineralized water~~ into the reactor vessel.

LA2

LA2

This test checks explosion of the charge associated with the tested loop, proper operation of the valves, and pump operability.

LA3

Replacement charges shall be selected such that the age of charge in service shall not exceed five years from the manufacturer's assembly date.

SR 3.1.7.7 d. Both systems, including both explosive valves, shall be tested in the course of two operating cycles.

LA2



~~3.4/4.4 STANDBY LIQUID CONTROL SYSTEM~~ (A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.4.B. Operation with Inoperable Components~~ (A1)

ACTION
A

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled and continued operation permitted provided that the component is returned to an operable condition within seven days.

~~4.4.B. Surveillance with Inoperable Components~~

- (L3)
1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

~~3.4.C Sodium Pentaborate Solution~~ (A1)

LCO
3.1.7

At all times when the Standby Liquid Control System is required to be OPERABLE, the following conditions shall be met:

SR 3.1.7.4

1. At least 186 pounds Boron-10 must be stored in the Standby Liquid Control Solution Tank and be available for injection.

SR 3.1.7.3

2. The sodium pentaborate solution concentration must be equal to or less than 9.2% by weight.

(L2) OR
Verify the concentration and temperature of boron in solution are within the limits of Figure 3.1.7-1

(L2) Add Figure 3.1.7-1 →

~~4.4.C Sodium Pentaborate Solution~~ (A1)

The following tests shall be performed to verify the availability of the Liquid Control Solution:

SR 3.1.7.1

1. Volume: Check at least once per day.

SR 3.1.7.2

2. Sodium Pentaborate Concentration check at least once per month. Also check concentration within 24 hours anytime water or boron is added to the solution.

SR 3.1.7.4

3. Boron-10 Quantity:

At least once per month, calculate and record the quantity of Boron-10 stored in the Standby Liquid Control Solution Tank.

SR 3.1.7.9

4. Boron-10 Enrichment: At least once per 18 months and following each addition of boron to the Standby Liquid Control Solution Tank:

(M1)

Proposed
SR 3.1.7.2 →



~~3.4/4.4 STANDBY LIQUID CONTROL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

SR 3.1.7.9

- a. Calculate the enrichment within 24 hours.
- b. Verify by analysis within 30 days.

(A1)

3.4.D ~~Standby Liquid Control System Requirements~~

4.4.D ~~Standby Liquid Control System Requirements~~

SR 3.1.7.5

The Standby Liquid Control System conditions must satisfy the following equation.

$$\frac{(C)(Q)(E)}{(13 \text{ wt.}\%)(86 \text{ gpm})(19.8 \text{ atom}\%)} \geq 1$$

where,

C = sodium pentaborate solution concentration (weight percent)

LA2 - Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.2.

Q = pump flow rate (gpm)

LA2 - Determined by the most recent performance of the surveillance instruction required by Specification 4.4.A.2.b

E = Boron-10 enrichment (atom percent Boron-10)

LA2 - Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.4.

ACTIONS B+C

1. If Specification 3.4.A through 3.4.D cannot be met, make at least one subsystem OPERABLE within 8 hours or the reactor shall be placed in a SHUTDOWN CONDITION with all operable control rods fully inserted within the following 12 hours.

SR 3.1.7.5

Verify that the equation given in Specification 3.4.D is satisfied at least once per month and within 24 hours anytime water or boron is added to the solution.

(A1)

1. No additional surveillance required.



UNIT 2
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

~~3.4/4.4 STANDBY LIQUID CONTROL SYSTEM~~

NOV 22 1988

~~LIMITING CONDITIONS FOR OPERATION~~

(AI)

~~SURVEILLANCE REQUIREMENTS~~

3.4 ~~STANDBY LIQUID CONTROL SYSTEM~~

Applicability

Applies to the operating status of the Standby Liquid Control System.

Objective

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

Specification

4.4 ~~STANDBY LIQUID CONTROL SYSTEM~~

Applicability

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective

To verify the operability of the Standby Liquid Control System.

Specification

(AI) A. ~~Normal System Availability~~

- 1. Except as specified in 3.4.B.1, the Standby Liquid Control System shall be OPERABLE at all times when there is fuel in the reactor vessel and the reactor is not in a shutdown condition with Specification 3.3.A.1 satisfied.

LCO 3.1.7

(LI)

(AI) A. ~~Normal System Availability~~

The operability of the Standby Liquid Control System shall be verified by the performance of the following tests:

- 1. Verify pump OPERABILITY in accordance with Specification 1.0.MM.
- 2. At least once during each operating cycle:

(P2)

- a. Check that the seating of the system relief valves is $1,425 \pm 75$ psig.

(LAI)

SR 3.1.7.8

- b. Manually initiate the system, except explosive valves. Visually verify flow by pumping boron solution through the recirculation path and back to the Standby Liquid Control Solution Tank. After pumping boron solution, the system shall be flushed with demineralized water. Verify minimum

(LAZ)

SR 3.1.7.6

~~3.4/4.4 STANDBY LIQUID CONTROL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A) ~~4.4.4 Normal System Applicability~~

~~4.4.4.2.b. (Cons'd)~~

SR 3.1.7.6 - pump flow rate of 39 gpm against a system head of 1275 psig by pumping demineralized water from the Standby Liquid Control Test Tank.

LA2

SR 3.1.7.7 c. ~~Manually~~ initiate one of the Standby Liquid Control System loops and pump demineralized water into the reactor vessel.

LA2

This test checks explosion of the charge associated with the tested loop, proper operation of the valves, and pump operability.

LA2

Replacement charges shall be selected such that the age of charge in service shall not exceed five years from the manufacturer's assembly date.

LA3

SR 3.1.7.7 d. Both systems, including both explosive valves shall be tested in the course of two operating cycles.

LA2

FEB 28 1995

~~3.4/4.4 STANDBY LIQUID CONTROL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

(AI)

~~SURVEILLANCE REQUIREMENTS~~

~~3.4.B. Operation with Inoperable Components~~

(AI)

~~4.4.B. Surveillance with Inoperable Components~~

ACTION A

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled and continued operation permitted provided that the component is returned to an operable condition within seven days.

(L3)

1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

~~3.4.C Sodium Pentaborate Solution~~

(AI)

~~4.4.C Sodium Pentaborate Solution~~

(AI)

LCO 3.1.7

(AI)

At all times when the Standby Liquid Control System is required to be OPERABLE, the following conditions shall be met:

The following tests shall be performed to verify the availability of the Liquid Control Solution:

SR 3.1.7.4

SR 3.1.7.3

1. At least 186 pounds Boron-10 must be stored in the Standby Liquid Control Solution Tank and be available for injection.
2. The sodium pentaborate solution concentration must be equal to or less than 9.2% by weight.

SR 3.1.7.1

SR 3.1.7.3

SR 3.1.7.4

SR 3.1.7.9

1. Volume: Check at least once per day.
2. Sodium Pentaborate Concentration check at least once per month. Also check concentration within 24 hours anytime water or boron is added to the solution.
3. Boron-10 Quantity:
At least once per month, calculate and record the quantity of Boron-10 stored in the Standby Liquid Control Solution Tank.
4. Boron-10 Enrichment: At least once per 18 months and following each addition of boron to the Standby Liquid Control Solution Tank:

OR

(L2)

Verify the concentration and temperature of boron in solution are within the limits of Figure 3.1.7-1

(L2)

Add Figure 3.1.7-1

(M.I)

Proposed SR 3.1.7.2



DEC 07 1994

~~3.4/4.4 STANDBY LIQUID CONTROL SYSTEM~~

~~EMITTING CONDITIONS FOR OPERATION~~

(AI)

~~SURVEILLANCE REQUIREMENTS~~

SR 3.1.7.9

- a. Calculate the enrichment within 24 hours.
- b. Verify by analysis within 30 days.

(AI)

3.4.D ~~Standby Liquid Control System Requirements~~

SR 3.1.7.5

The Standby Liquid Control System conditions must satisfy the following equation.

$$\frac{(C)(Q)(E)}{(13 \text{ wt.}\%)(86 \text{ gpm})(19.8 \text{ atom}\%)} \geq 1$$

where,

C = sodium pentaborate solution concentration (weight percent)

(LAZ)

Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.2.

Q = pump flow rate (gpm)

(LAZ)

Determined by the most recent performance of the surveillance instruction required by Specification 4.4.A.2.b.

E = Boron-10 enrichment (atom percent Boron-10)

(LAZ)

Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.4.

ACTIONS B+C

1. If Specification 3.4.A through 3.4.D cannot be met, make at least one subsystem OPERABLE within 8 hours or the reactor shall be placed in a SHUTDOWN CONDITION with all OPERABLE control rods fully inserted within the following 12 hours.

4.4.D ~~Standby Liquid Control System Requirements~~

SR 3.1.7.5

Verify that the equation given in Specification 3.4.D is satisfied at least once per month and within 24 hours anytime water or boron is added to the solution.

(AI)

1. No additional surveillance required.



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



NOV 22 1988

~~3.4/4.4 STANDBY LIQUID CONTROL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the operating status of the Standby Liquid Control System.

Objective

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

Specification

A. ~~Normal System Availability~~

1. Except as specified in 3.4.B.1, the Standby Liquid Control System shall be OPERABLE at all times when there is fuel in the reactor vessel and the reactor is not in a shutdown condition with Specification 3.3.A.1 satisfied.

LCD 3.1.7

(L1)

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective

To verify the operability of the Standby Liquid Control System.

Specification

A. ~~Normal System Availability~~

(A1) The operability of the Standby Liquid Control System shall be verified by the performance of the following tests:

1. Verify pump operability in accordance with Specification 1.0.MM.

(A2)

2. At least once during each operating cycle:

(LA1)

- a. Check that the setting of the system relief valves is $1,425 \pm 75$ psig.

SR 3.1.7.8

(LA2)

- b. Manually initiate the system, except explosive valves. Visually verify flow by pumping boron solution through the recirculation path and back to the Standby Liquid Control Solution Tank. After pumping boron solution, the system shall be flushed with demineralized water.

SR 3.1.7.6

JW



SEP 02 1968

~~3.4/4.4 STANDBY LIQUID CONTROL SYSTEM~~

(A1)

LIMITING CONDITIONS FOR OPERATION

~~SURVEILLANCE REQUIREMENTS~~

~~4.4.A Normal System Availability (Cont'd)~~

(A1)

~~4.4.A.2.b. (Cont'd)~~

SR 3.1.7.6

pump flow rate of 39 gpm against a system head of 1275 psig by pumping demineralized water from the Standby Liquid Control Test Tank.

(LA2)

SR 3.1.7.7 c. Manually initiate one of the Standby Liquid Control System loops and pump demineralized water into the reactor vessel.

(LA2)

This test checks explosion of the charge associated with the tested loop, proper operation of the valves, and pump operability.

(LA2)

Replacement charges shall be selected such that the age of charge in service shall not exceed five years from the manufacturer's assembly date.

(LA3)

SR 3.1.7.7 d. Both systems, including both explosive valves, shall be tested in the course of two operating cycles.

(LA2)



FEB 28 1995

~~2.4/4.4 STANDBY LIQUID CONTROL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.4.B. Operation with Inoperable Components~~

(A1)

~~4.4.B. Surveillance with Inoperable Components~~

ACTION A

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled and continued operation permitted provided that the component is returned to an operable condition within seven days.

(L3)

1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

~~3.4.C Sodium Pentaborate Solution~~

(A1)

~~4.4.C Sodium Pentaborate Solution~~

(A1)

LCO 3.1.7

At all times when the Standby Liquid Control System is required to be OPERABLE, the following conditions shall be met:

The following tests shall be performed to verify the availability of the Liquid Control Solution:

SR 3.1.7.4

1. At least 186 pounds Boron-10 must be stored in the Standby Liquid Control Solution Tank and be available for injection.

SR 3.1.7.1

1. Volume: Check at least once per day.

SR 3.1.7.3

2. The sodium pentaborate solution concentration must be equal to or less than 9.2% by weight.

SR 3.1.7.3

2. Sodium Pentaborate Concentration check at least once per month. Also check concentration within 24 hours anytime water or boron is added to the solution.

SR 3.1.7.4

3. Boron-10 Quantity:

At least once per month, calculate and record the quantity of Boron-10 stored in the Standby Liquid Control Solution Tank.

(L2)

OR
Verify the concentration and temperature of boron in solution are within the limits of Figure 3.1.7-1

SR 3.1.7.9

4. Boron-10 Enrichment: At least once per 18 months and following each addition of boron to the Standby Liquid Control Solution Tank:

(L2)

ADD Figure 3.1.7-1

(M1)

Proposed SR 3.1.7.2



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

(A1)

3.4.D Standby Liquid Control System Requirements

4.4.D Standby Liquid Control System Requirements

SR 3.1.7.5

SR 3.1.7.5

The Standby Liquid Control System conditions must satisfy the following equation.

Verify that the equation given in Specification 3.4.D is satisfied at least once per month and within 24 hours anytime water or boron is added to the solution.

$$\left(\frac{C}{13 \text{ wt.}\%}\right)\left(\frac{Q}{86 \text{ gpm}}\right)\left(\frac{E}{19.8 \text{ atom}\%}\right) \geq 1$$

where,

C = sodium pentaborate solution concentration (weight percent)

(LA2)

~~Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.2.~~

Q = pump flow rate (gpm)

(LA2)

~~Determined by the most recent performance of the surveillance instruction required by Specification 4.4.A.2.b.~~

E = Boron-10 enrichment (atom percent Boron-10)

(LA2)

~~Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.4.~~

Actions B+C

- If Specification 3.4.A through 3.4.D cannot be met, make at least one subsystem OPERABLE within 8 hours or the reactor shall be placed in a SHUTDOWN CONDITION with all OPERABLE control rods fully inserted within the following 12 hours.

(A1)

- ~~No additional surveillance required.~~



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.7 - STANDBY LIQUID CONTROL SYSTEM**

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specification, NUREG 1433. These changes should make the BFN Technical Specifications easier for the operator (and other users) to read and understand. During the reformatting and renumbering process, no technical changes (either actual or interpretational) were made unless they were identified and justified.
- A2 Surveillance Requirements for pump operability that are required by the Inservice Testing (IST) Program have been removed from individual Specifications. This change is considered administrative in nature since these requirements remain in the IST Program which is defined by proposed Specification 5.5.6.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 Added Surveillance to verify the continuity of the explosive charge. The continuity check is intended to ensure proper operation will occur if required.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 Verification of the relief valve's proper operation and setpoint is conducted in accordance with the plant's Inservice Test Program and the ASME code.
- LA2 The method of performing surveillance tests is relocated to plant procedures.
- LA3 Requirements on the replacement charges for explosive valves have been relocated to the Bases and plant administrative controls.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.7 - STANDBY LIQUID CONTROL SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (CONTINUED)

"Specific"

- L1 The CTS states applicability is at all times when fuel is in the vessel and the reactor is not in a shutdown condition with BFN TS 3.3.A.1 satisfied. The proposed ISTS Specification does not require SLC System operability during Hot Shutdown, Cold Shutdown, or Refueling (Modes 3, 4, & 5) since control rod withdrawal is limited and adequate SDM prevents criticality under these conditions.
- L2 Added the second part of SR 3.1.7.3, which provides the flexibility of allowing the concentration of boron in solution to be greater than 9.2% by weight as long as it is within the limits of proposed Figure 3.1.7-1 and the equation of SR 3.1.7.5 is met. Figure 3.1.7-1 has been added to allow this flexibility. This is acceptable since there is a 10°F thermal margin to unwanted precipitation of the sodium pentaborate. Per FSAR Chapter 3.8.3, the worst case sodium pentaborate solution concentration required to shutdown the reactor with sufficient margin to account for 0.05 $\Delta k/k$ and Xenon poisoning effects is 9.2 weight percent. This corresponds to a 40°F saturation temperature. The worst case SLCS equipment area temperature is not predicted to fall below 50°F. SR 3.1.7.3 must be performed within 8 hours of discovery that the concentration is > 9.2 weight percent and every 12 hours thereafter until the concentration is verified \leq 9.2 weight percent. This Frequency is appropriate under these conditions taking into consideration the SLC System design capability still exists for vessel injection and the low probability of the temperature and concentration limits of Figure 3.1.7-1 not being met.
- L3 Deleted BFN TS 4.4.B.1, which requires that when a component is found inoperable, its redundant component be demonstrated operable immediately and daily thereafter until the inoperable component is repaired. This requirement is deleted for several reasons. Increased testing has not been shown to demonstrate operability any better than testing at the normal SR test interval. In many cases, increased testing adds to the failure rates of components by increasing wear and tear. Common mode failure analysis in conjunction with loss of function analyses provide adequate assurance of redundant system operability. Loss of function determination program controls are provided by BFN ISTS 5.5.11.

UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.3.E Reactivity Control

If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within 24 hours.

4.3.E Reactivity Control

Surveillance requirements are as specified in 4.3.C and .D above.

See Justification for Changes for BFN 1575 3.1.2 + 3.1.4

(A1) ~~3.3.F Scram Discharge Volume (SDV)~~

(L1) Proposed Note at Start of Action Table

1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.

(L2)

(M1)

ACTION A

2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.

(L1)

ACTION 3. B If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.

Action C

(M2)

Shutdown

(A1) ~~4.3.F Scram Discharge Volume (SDV)~~

SR 3.1.8.1 1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.

(A2)

SR 3.1.8.1 Note

(A5)

SR 3.1.8.2 1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.

(A3)

2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.

(L3)

3. No additional surveillance required.

(A4)

(M3)

Proposed SR 3.1.8.3

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



DEC 07 1994

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.3.E. Reactivity Control

If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within 24 hours.

4.3.E. Reactivity Control

Surveillance requirements are as specified in 4.3.C and .D above.

See Justification for Changes for BFN ISTS 3.1.2 + 3.1.4

(A1) ~~3.3.F. Scram Discharge Volume (SDV)~~
L1 Proposed Note at start of ACTIONS Table

LLO 3.1.8
1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.

L2

M1

ACTION A

2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.

ACTION B 3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.
ACTION C

M2 12 Shutdown

(A1) ~~4.3.F. Scram Discharge Volume (SDV)~~
SR 3.1.8.1

1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.

A2

SR 3.1.8.1 Note

A5

SR 3.1.8.2 1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.

A3

2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.

L3

A4 3. No additional surveillance required.

M3 Proposed SR 3.1.8.3

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



DEC 07 1994

~~3.3/4.3 REACTIVITY CONTROL~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

3.3.E. Reactivity Control

If Specifications 3.3.C and 3.3.D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within 24 hours.

4.3.E. Reactivity Control

Surveillance requirements are as specified in 4.3.C and 4.3.D above.

~~3.3.F. Scram Discharge Volume (SDV)~~

Proposed Note at Start of ACTION Table

1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.

LCO 3.1.8

(L2)

(M1)

Action A

2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.

(L1)

3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.

Action B

Action C

(M2)

(L2)

Shutdown

(A1)

~~4.3.F. Scram Discharge Volume (SDV)~~

See Justification for Change for BFN ISTS 3.1.2 + 3.1.4

SR3.1.8.1

1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter.

(A2)

SR 3.1.8.1 Note

The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.

(A5)

SR3.1.8.2

1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.

(A3)

2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.

(L3)

3. No additional surveillance required.

(A4)

(M3)

Proposed SR 3.1.8.3



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.8 - SDV VENT AND DRAIN VALVES**

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering is in accordance with the BWR/4 Standard Technical Specifications (STS), NUREG-1433. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the improved Technical Specifications, no technical changes (either actual or interpretational) to the TS were made unless they were identified and justified.
- A2 CTS Surveillance Requirement 4.3.F.1.a requires that the SDV drain and vent valves be verified open PRIOR TO STARTUP. These words are unnecessary and were deleted to make the BFN ISTS SR 3.1.8.1 consistent with the BWR Standard Technical Specifications, NUREG-1433. Proposed SR 3.1.8.1 requires the valves to be verified open when they are required to be operable in Modes 1 and 2. Proposed SR 3.0.4 does not allow entry into a Mode unless the SRs have been met within their specified frequency. Therefore, this SR is required to be met prior to entry into Mode 2 or "prior to startup." Since the intent of the SR is not changed, the deletion of these words are considered administrative.
- A3 CTS 4.3.F.1.b requires the SDV drain and vent valves to be demonstrated OPERABLE in accordance with Specification 1.0.MM, which is the Surveillance Requirements for ASME Section XI Pump and Valve Program. This program provides equivalent testing requirements, with respect to valve cycling not closure times, to proposed SR 3.1.8.2, which requires each SDV vent and drain valve to be cycled fully closed and open every 92 days. Therefore, the proposed change is considered administrative.
- A4 Deleted CTS 4.3.F.3, which states no additional surveillance required, to make the BFN ISTS consistent with NUREG-1433. It is unnecessary to specify that no additional surveillance is required - omission of this statement would serve the same purpose. Therefore, the proposed change is considered administrative.
- A5 The Note in proposed SR 3.1.8.1 provides an allowance that does not require the surveillance to be met on SDV vent and drain valves that are closed during the performance of SR 3.1.8.2, which requires valves to be cycled fully closed and open every 92 days. CTS allow the valves to be closed intermittently for testing but this is not allowed to exceed 1 hour in any 24-hour period during operation. Since each SDV vent and drain valve is required to close in ≤ 60 seconds per proposed SR 3.1.8.3, the current 1 hour allowance for the valves to be closed for



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.8 - SDV VENT AND DRAIN VALVES**

ADMINISTRATIVE CHANGES (CONTINUED)

testing in any 24-hour period will not be exceeded when cycling the valves to the fully closed and fully open position. Since the intent is the same (i.e., to allow the SDV vent and drain valves to be cycled during reactor operations), the proposed change is considered to be administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 CTS 3.3.F allows unlimited continued operation when any SDV drain and vent valve becomes inoperable provided that the redundant drain or vent valve is demonstrated OPERABLE immediately and weekly thereafter. Proposed Action A is more restrictive since it allows continued operation for 7 days. At that time if the valve has not been restored to OPERABLE status, the reactor must be placed in MODE 3 within 12 hours.
- M2 Proposed Action C requires the plant to be in MODE 3 in 12 hours while CTS 3.3.F.3 requires the plant to be in HOT STANDBY CONDITION (equivalent to MODE 2 at $\leq 1\%$ RTP) within 24 hours of redundant drain or vent valves becoming inoperable. Proposed Action C is more restrictive since it does not allow as much time to change modes and requires the reactor to be placed in MODE 3 versus HOT STANDBY (equivalent to MODE 2 at $\leq 1\%$ RTP of proposed BFN ISTS).
- M3 Added SR 3.1.8.3, which requires that an integrated test of the SDV vent and drain valves be performed on an 18 month frequency to verify total system performance. After the receipt of a simulated or actual scram and subsequent scram reset signal, the closure and subsequent opening of the SDV vent and drain valves, respectively, are verified. The closure time of 60 seconds is acceptable based on the bounding leakage for release of reactor coolant outside containment. The LOGIC SYSTEM FUNCTIONAL TEST in Proposed LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3 overlap this Surveillance to provide complete testing of the assumed safety function.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.8 - SDV VENT AND DRAIN VALVES

TECHNICAL CHANGES - LESS RESTRICTIVE

- L1 Added a proposed Note ("Separate Condition entry is allowed for each SDV vent and drain line") at the start of the ACTIONS Table to provide more explicit instructions for proper application of the Actions for Technical Specification compliance. In conjunction with the proposed Specification 1.3- "Completion Times," this Note provides direction consistent with the intent of the proposed Actions for inoperable SDV vent and drain valves. Each SDV line is intended to be allowed a specified period of time to confirm it isolated or is capable of isolation, and to restore the complete function of the line.

Current TS 3.3.F.3 requires the reactor to be in Hot Standby Condition within 24 hours if both valves are inoperable in one or more SDV vent or drain lines. Proposed Action B allows 8 hours to isolate the line(s). Both valves must be restored to operable status within 7 days per Action A. Recognizing that the SDV vent and drain valves are normally open to prevent accumulation of water in the SDV from leakage, a Note has been added to Required Action B.1 (which requires isolation of the line), allowing periodic opening of the affected line for draining and venting the SDV. This may be necessary due to CRD seal leakage in order to avoid automatic reactor scrams on high level in the SDV. These extended times, and the option to administratively un-isolate a SDV line isolated by a Required Action, are consistent with the BWR Standard Technical Specifications, NUREG 1433. These increased allowances are deemed not to substantially increase the risk of a scram with an additional failure that could allow the SDV to remain un-isolated; nor to substantially increase the risk of the SDV failing to accept the control rod drive water displaced during a scram.

- L2 CTS 3.F.1 requires the SDV drain and vent valves to be OPERABLE any time that the reactor protection system (RPS) is required to be OPERABLE. Proposed BFN ISTS 3.1.8 requires the SDV vent and drain valves to be OPERABLE in Modes 1 and 2. Currently, portions of the RPS are required to be OPERABLE during other MODES, as described in BFN TS Table 3.1.A, therefore, the proposed Specification is considered less restrictive. The proposed Specification applicability is based on when a full scram may be required. In MODES 3 and 4, control rods are only allowed to be withdrawn under proposed Special Operations LCO 3.10.3 and 3.10.4, which provide adequate controls to ensure that only a single control rod can be withdrawn. Also, during MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. The SDV vent and drain valves need not be OPERABLE in these MODES since the reactor is

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.1.8 - SDV VENT AND DRAIN VALVES**

TECHNICAL CHANGES - LESS RESTRICTIVE (CONTINUED)

subcritical, only one rod may be withdrawn, and the SDV is adequate to contain the water from the single rod scram even if isolated.

- L3 Deleted BFN TS 4.3.F.2, which requires that when a component is found inoperable, its redundant component be demonstrated operable immediately and daily thereafter until the inoperable component is repaired. This requirement is deleted for several reasons. Increased testing has not been shown to demonstrate operability any better than testing at the normal SR test interval. In many cases, increased testing adds to the failure rates of components by increasing wear and tear. Common mode failure analysis in conjunction with loss of function analyses provide adequate assurance of redundant system operability. Loss of function determination program controls are provided by BFN ISTS 5.5.11.



UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

MAY 20 1993

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.5.P Average Planar Linear Heat Generation Rate

4.5.I Average Planar Linear Heat Generation Rate (APLHGR)

(A2) During steady-state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of any fuel assembly at any axial location shall not exceed the appropriate APLHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the ~~COLD SHUTDOWN CONDITION~~ within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

ACTION A

ACTION B

(A3)

Applicability

LCO 3.2.1

SR 3.2.1.1

The APLHGR shall be checked daily during reactor operation at \geq 25% rated thermal power.

(M1) Add proposed first Frequency of SR 3.2.1.1

(L1)

(M2) \geq 25% RTP within 4 hours

J. Linear Heat Generation Rate (LHGR)

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT.

J. Linear Heat Generation Rate (LHGR)

The LHGR shall be checked daily during reactor operation at \geq 25% rated thermal power.

See Justification for Changes for BFN 1STS 3.2.3

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

MAY 20 1993

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.I Average Planar Linear Heat Generation Rate A1

Applicability

A2 During steady-state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of any fuel assembly at any axial location shall not exceed the appropriate APLHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, Action A action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the ~~COLD SHUTDOWN CONDITION~~ within 36 hours. LI

Action B

A3 Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

LCO
3.2.1

4.5.I Average Planar Linear Heat Generation Rate (APLHGR)

SR 3.2.1.1

The APLHGR shall be checked daily during reactor operation at $\geq 25\%$ rated thermal power.

M1 Add proposed first Frequency of SR 3.2.1.1

M2 $\geq 25\%$ RTP within 4 hours

J. Linear Heat Generation Rate (LHGR)

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and

J. Linear Heat Generation Rate (LHGR)

The LHGR shall be checked daily during reactor fuel operation at $\geq 25\%$ rated thermal power.

See Justification for Changes for BFN 15TS 3.2.3



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

MAY 20 1993

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.1 Average Planar Linear Heat Generation Rate

(A1)

(A2) During steady-state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of any fuel assembly at any axial location shall not exceed the appropriate APLHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

LCO 3.2.1

(L1)

(A3)

ACTION A

ACTION B

Applicability

4.5.1 Average Planar Linear Heat Generation Rate (APLHGR)

SR 3.2.1.1

The APLHGR shall be checked daily during reactor operation at $\geq 25\%$ rated thermal power.

(M1) Add proposed first Frequency of SR 3.2.1.1

(M2) $< 25\%$ RTP within 4 hours

J. Linear Heat Generation Rate (LHGR)

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT.

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

J. Linear Heat Generation Rate (LHGR)

The LHGR shall be checked daily during reactor operation at $\geq 25\%$ rated thermal power.

See Justification for Changes for BFN 15TS 3.2.3



JUSTIFICATION FOR CHANGES
BFN ISTS 3.2.1 - AVERAGE PLANAR LINEAR HEAT GENERATION RATE

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specification, NUREG 1433. These changes should make the BFN Technical Specifications easier for the operator (and other users) to read and understand. During the reformatting and renumbering process, no technical changes (either actual or interpretational) were made unless they were identified and justified.
- A2 The Applicability has been changed from "steady-state power operation" to "Thermal Power \geq 25% RTP." This change is considered administrative in nature since, based on a CTS surveillance frequency of daily during reactor operation at \geq 25% rated thermal power, the intent of CTS is the same as the proposed ISTS specification.
- A3 The requirement to continue the surveillance when the limits are not met has been deleted since the total allowed completion time for restoring the limit or placing the plant in a condition outside the Applicability is 6 hours. Since the 6 hour time frame is less than the Surveillance Frequency of 24 hours, the surveillance would not be required to be performed again while the plant was in the action. The requirement to continue to comply with actions until the limits are met has been moved and is now addressed by LCO 3.0.2, which clarifies that if an LCO is met or is no longer applicable prior to the expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated. As a result, these changes are considered administrative in nature.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 A new Frequency has been added to require verification of APLHGR within 12 hours of reaching or exceeding 25% RTP. This is an additional restriction on plant operation. CTS would allow up to 24 hours after reaching 25% RTP to perform the test.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.2.1 - AVERAGE PLANAR LINEAR HEAT GENERATION RATE

- M2 The requirement to place the plant in a COLD SHUTDOWN CONDITION within 36 hours when the limit is not restored within the required completion time has been revised to reflect placing the plant in a non-applicable condition. Current Specification 1.O.C.1 states action requirements are applicable during the operational conditions of each specification. Therefore, the requirement to place the plant in cold shutdown is not applicable after thermal power is reduced below 25% RTP. The revised Action requires plant power to be reduced to < 25% RTP (outside the applicable condition) within 4 hours. The current action allows 36 hours to place the plant in a non-applicable condition. As such, this is an additional restriction on plant operation.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Specific"

- L1 The requirement to initiate action within 15 minutes to restore the limit is relaxed and relocated to the Bases in the form of a discussion that "prompt action" should be taken to restore the parameter to within limits. Immediate action may not always be the conservative method to assure safety. The 2 hour completion time for restoration of the limit allows appropriate actions to be evaluated by the operator and completed in a timely manner.



UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.5.J Linear Heat Generation Rate (LHGR)

3.5.J (Cont'd)

If at any time during steady-state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.K Minimum Critical Power Ratio (MCPR)

LCO
3.2.2

(A2)

Applicability

ACTION
A

ACTION
B

(A3)

The minimum critical power ratio (MCPR) shall be equal to or greater than the operating limit MCPR (OLMCPR) as provided in the CORE OPERATING LIMITS REPORT.

If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5.J Linear Heat Generation Rate (LHGR)

See Justification for Change for BFN ISTS 3.2.3

(M1)

Add proposed 1st Frng. of SR 3.2.2.1

4.5.K Minimum Critical Power Ratio (MCPR)

SR 3.2.2.1

1. MCPR shall be checked daily during reactor power operation at $\geq 25\%$ rated thermal power and following any change in power level or distribution that would cause operation with a LIMITING CONTROL ROD PATTERN.

SR 3.2.2.2

2. The MCPR limit at rated flow and rated power shall be determined as provided in the CORE OPERATING LIMITS REPORT using:

- a. as defined in the CORE OPERATING LIMITS REPORT prior to initial scram time measurements for the cycle, performed in accordance with Specification 4.3.C.1.

BFN
Unit 1

3.5/4.5-19

AMENDMENT NO. 216

(M2) < 25% RTP within 4 hours



FEB 24 1995

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.K Minimum Critical Power Ratio (MCPR)~~

~~4.5.K Minimum Critical Power Ratio (MCPR)~~

~~4.5.K.2 (Cont'd)~~

b. τ as defined in the CORE OPERATING LIMITS REPORT following the conclusion of each scram-time surveillance test required by Specifications 4.3.C.1 and 4.3.C.2.

(LAI)

SR 3.2.2.2

The determination of the limit must be completed within 72 hours of each scram-time surveillance required by Specification 4.3.C.

L. APRM Setpoints

1. Whenever the core thermal power is $\geq 25\%$ of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0 , or the APRM scram setpoint equation listed in Section 2.1.A and the APRM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT shall be multiplied by FRP/CMFLPD.
2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.
3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to $\leq 25\%$ of rated thermal power within 4 hours.

L. APRM Setpoints

FRP/CMFLPD shall be determined daily when the reactor is $\geq 25\%$ of rated thermal power.

(see justification for changes for BFN ISTS 3.2.4)



UNIT 2
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

DEC 07 1994

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

J. Linear Heat Generation Rate (LHGR)

3.5.J (Cont'd)

corresponding action shall continue until reactor operation is within the prescribed limits.

~~3.5.K Minimum Critical Power Ratio (MCPR)~~

LCO 3.2.2

The minimum critical power ratio (MCPR) shall be equal to or greater than the operating limit MCPR (OLMCPR) as provided in the CORE OPERATING LIMITS REPORT.

(A2)

Applicability

If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the GOLD SHUTDOWN CONDITION within 36 hours, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

ACTION A

ACTION B

(A3)

(M2)

< 25% RTP within 4 hours

J. Linear Heat Generation Rate (LHGR)

See Justification for Changes for BFN ISTS 3.2.3

Add proposed 1st Frequency of SR 3.2.2.1 (M1)

~~3.5.K Minimum Critical Power Ratio (MCPR)~~

SR 3.2.2.1

1. MCPR shall be checked daily during reactor power operation at > 25% rated thermal power and following any change in power level or distribution that would cause operation with a LIMITING CONTROL ROD PATTERN.

(L2)

SR 3.2.2.2

2. The MCPR limit at rated flow and rated power shall be determined as provided in the CORE OPERATING LIMITS REPORT using:

(LAI)

a. \bar{v} as defined in the CORE OPERATING LIMITS REPORT prior to initial scram time measurements for the cycle, performed in accordance with Specification 4.3.C.1.

b. \bar{v} as defined in the CORE OPERATING LIMITS REPORT following the conclusion of each scram-time surveillance test required by Specifications 4.3.C.1 and 4.3.C.2.

SR 3.2.2.2

The determination of the limit must be completed within 72 hours of each scram-time surveillance required by Specification 4.3.C.

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.K Minimum Critical Power Ratio (MCPR)~~

~~4.5.K Minimum Critical Power Ratio (MCPR)~~

LCO 3.2.2
A2
Applicable
The minimum critical power ratio (MCPR) shall be equal to or greater than the operating limit MCPR (OLMCPR) as provided in the CORE OPERATING LIMITS REPORT. If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, (L1)

ACTION A
action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. (A3)

ACTION B
M2
< 25% RTP within 4 hours

(L1)

SR 3.2.2.1
1. MCPR shall be checked daily during reactor power operation at $\geq 25\%$ rated thermal power and following any change in power level or distribution that would cause operation with a LIMITING CONTROL ROD PATTERN. (L2)

SR 3.2.2.2
2. The MCPR limit at rated flow and rated power shall be determined as provided in the CORE OPERATING LIMITS REPORT using:

a. T as defined in the CORE OPERATING LIMITS REPORT prior to initial scram time measurements for the cycle, performed in accordance with Specification 4.3.C.1. (LAI)

b. T as defined in the CORE OPERATING LIMITS REPORT following the conclusion of each scram-time surveillance test required by Specifications 4.3.C.1 and 4.3.C.2.

SR 3.2.2.2
The determination of the limit must be completed within 72 hours of each scram-time surveillance required by Specification 4.3.C.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.2.2 - MINIMUM CRITICAL POWER RATIO

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specification, NUREG 1433. These changes should make the BFN Technical Specifications easier for the operator (and other users) to read and understand. During the reformatting and renumbering process, no technical changes (either actual or interpretational) were made unless they were identified and justified.
- A2 The Applicability has been changed from "steady-state operation" to "Thermal Power \geq 25% RTP." This change is considered administrative in nature since, based on a CTS surveillance frequency of daily during reactor operation at \geq 25% rated thermal power, the intent of CTS is the same as the proposed ISTS specification.
- A3 The requirement to continue the surveillance when the limits are not met has been deleted since the total allowed completion time for restoring the limit or placing the plant in a condition outside the Applicability is 6 hours. Since the 6 hour time frame is less than the Surveillance Frequency of 24 hours, the surveillance would not be required to be performed again while the plant was in the action. The requirement to continue to comply with actions until the limits are met has been moved and is now addressed by LCO 3.0.2, which clarifies that if an LCO is met or is no longer applicable prior to the expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated. As a result, these changes are considered administrative in nature.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 A new Frequency has been added to require verification of MCPR within 12 hours of reaching or exceeding 25% RTP. This is an additional restriction on plant operation. CTS would allow up to 24 hours after reaching 25% RTP to perform the test.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.2.2 - MINIMUM CRITICAL POWER RATIO

- M2 The requirement to place the plant in a COLD SHUTDOWN CONDITION within 36 hours when the limit is not restored within the required completion time has been revised to reflect placing the plant in a non-applicable condition. Current Specification 1.0.C.1 states action requirements are applicable during the operational conditions of each specification. Therefore, the requirement to place the plant in cold shutdown is not applicable after thermal power is reduced below 25% RTP. The revised Action requires plant power to be reduced to < 25% RTP (outside the applicable condition) within 4 hours. The current action allows 36 hours to place the plant in a non-applicable condition. As such, this is an additional restriction on plant operation.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 The method used to determine τ is moved to the Bases in the form of a discussion (describing the ways to compute τ). This information is also contained in the Core Operating Limits Report (COLR). The proposed change does not change the intent of CTS.

"Specific"

- L1 The requirement to initiate action within 15 minutes to restore the limit is relaxed and relocated to the Bases in the form of a discussion that "prompt action" should be taken to restore the parameter to within limits. Immediate action may not always be the conservative method to assure safety. The 2 hour completion time for restoration of the limit allows appropriate actions to be evaluated by the operator and completed in a timely manner.
- L2 Since a limiting control rod pattern is currently defined as operating on a power distribution limit such as MCPR, the condition is extremely unlikely and the surveillance would almost never be required. In the CTS, determination that the plant is operating with a limiting control rod pattern would be found during performance of the daily SRs for thermal limits. If operating with a thermal limit in excess of CTS limits, proper actions are required to restore the plant to within limits. To ensure that the plant is restored to within limits, the SRs must be performed anyway, thus the additional SR frequency during limiting control rod pattern is not necessary.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



MAY 20 1993

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.5.I Average Planar Linear Heat Generation Rate

During steady-state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of any fuel assembly at any axial location shall not exceed the appropriate APLHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5.I Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR shall be checked daily during reactor operation at \geq 25% rated thermal power.

See Justification for Change for BFN 15TS 3.2.1

~~3. Linear Heat Generation Rate (LHGR)~~

(A2) Applicability

LCO 3.2.3

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT.

~~3. Linear Heat Generation Rate (LHGR)~~

SR 3.2.3.1

The LHGR shall be checked daily during reactor operation at \geq 25% rated thermal power.

(M1) Add proposed 1st Frequency of SR 3.2.3.1



FEB 24 1995

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.J Linear Heat Generation Rate (LHGR)~~

~~4.5.J Linear Heat Generation Rate (LHGR)~~

~~3.5.J (Cont'd)~~

ACTION A

If at any time during steady-state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours.

ACTION B

Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

A3

L1

< 25% RTP within 4 hours

M2

See Justification for Changes for BFN 15TS 3.2.2

~~3.5.K Minimum Critical Power Ratio (MCPR)~~

~~4.5.K Minimum Critical Power Ratio (MCPR)~~

The minimum critical power ratio (MCPR) shall be equal to or greater than the operating limit MCPR (OLMCPR) as provided in the CORE OPERATING LIMITS REPORT. If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

1. MCPR shall be checked daily during reactor power operation at $\geq 25\%$ rated thermal power and following any change in power level or distribution that would cause operation with a LIMITING CONTROL ROD PATTERN.
2. The MCPR limit at rated flow and rated power shall be determined as provided in the CORE OPERATING LIMITS REPORT using:
 - a. T_c as defined in the CORE OPERATING LIMITS REPORT prior to initial scram time measurements for the cycle, performed in accordance with Specification 4.3.C.1.



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

MAY 20 1993

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.5.1 Average Planar Linear Heat Generation Rate

During steady-state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of any fuel assembly at any axial location shall not exceed the appropriate APLHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5.1 Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR shall be checked daily during reactor operation at \geq 25% rated thermal power.

See Justification for Change for BFN ISTS 32.1

Linear Heat Generation Rate (LHGR)

Applicability

(A2) During steady-state power operation,

LCO 3.2.3

the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and

ACTION A

ACTION B

Linear Heat Generation Rate (LHGR)

SR 3.2.3.1

The LHGR shall be checked daily during reactor fuel operation at \geq 25% rated thermal power.

(M1) Add proposed 1st Frequency of SR 3.2.3.1

(L1)

(M2) $<$ 25% RTP within 4 hours

(A3)



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

DEC 07 1994

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~J. Linear Heat Generation Rate (LHGR)~~

~~J. Linear Heat Generation Rate (LHGR)~~

(A1) ~~3.5.J (Cont'd)~~

corresponding action shall continue until reactor operation is within the prescribed limits.

(A3)

3.5.K Minimum Critical Power Ratio (MCPR)

4.5.K Minimum Critical Power Ratio (MCPR)

The minimum critical power ratio (MCPR) shall be equal to or greater than the operating limit MCPR (OLMCPR) as provided in the CORE OPERATING LIMITS REPORT. If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

1. MCPR shall be checked daily during reactor power operation at $\geq 25\%$ rated thermal power and following any change in power level or distribution that would cause operation with a LIMITING CONTROL ROD PATTERN.

2. The MCPR limit at rated flow and rated power shall be determined as provided in the CORE OPERATING LIMITS REPORT using:

a. \bar{T} as defined in the CORE OPERATING LIMITS REPORT prior to initial scram time measurements for the cycle, performed in accordance with Specification 4.3.C.1.

b. \bar{T} as defined in the CORE OPERATING LIMITS REPORT following the conclusion of each scram-time surveillance test required by Specifications 4.3.C.1 and 4.3.C.2.

The determination of the limit must be completed within 72 hours of each scram-time surveillance required by Specification 4.3.C.

See Justification for Changes for BFN ISTS 3.2.2



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

MAY 20 1993

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.5.I Average Planar Linear Heat Generation Rate

During steady-state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of any fuel assembly at any axial location shall not exceed the appropriate APLHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5.I Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR shall be checked daily during reactor operation at \geq 25% rated thermal power.

See Justification for Changes for BFN 15TS 3.2.1

J. Linear Heat Generation Rate (LHGR)

Applicability

(A2)

LCO 3.2.3

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT.

ACTION A

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

ACTION B

J. Linear Heat Generation Rate (LHGR)

SR 3.2.3.1

The LHGR shall be checked daily during reactor operation at \geq 25% rated thermal power.

(M1)

Add proposed 1st Frequency of SR 3.2.3.1

(M2)

\leq 25% RTP within 4 hours

(A3)

JUSTIFICATION FOR CHANGES
BFN ISTS 3.2.3 - LINEAR HEAT GENERATION RATE

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specification, NUREG 1433. These changes should make the BFN Technical Specifications easier for the operator (and other users) to read and understand. During the reformatting and renumbering process, no technical changes (either actual or interpretational) were made unless they were identified and justified.
- A2 The Applicability has been changed from "steady-state power operation" to "Thermal Power \geq 25% RTP." This change is considered administrative in nature since, based on a CTS surveillance frequency of daily during reactor operation at \geq 25% rated thermal power, the intent of CTS is the same as the proposed ISTS specification.
- A3 The requirement to continue the surveillance when the limits are not met has been deleted since the total allowed completion time for restoring the limit or placing the plant in a condition outside the Applicability is 6 hours. Since the 6 hour time frame is less than the Surveillance Frequency of 24 hours, the surveillance would not be required to be performed again while the plant was in the action. The requirement to continue to comply with actions until the limits are met has been moved and is now addressed by LCO 3.0.2, which clarifies that if an LCO is met or is no longer applicable prior to the expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated. As a result, these changes are considered administrative in nature.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 A new Frequency has been added to require verification of LHGR within 12 hours of reaching or exceeding 25% RTP. This is an additional restriction on plant operation. CTS would allow up to 24 hours after reaching 25% RTP to perform the test.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.2.3 - LINEAR HEAT GENERATION RATE

- M2 The requirement to place the plant in a COLD SHUTDOWN CONDITION within 36 hours when the limit is not restored within the required completion time has been revised to reflect placing the plant in a non-applicable condition. Current Specification 1.O.C.1 states action requirements are applicable during the operational conditions of each specification. Therefore, the requirement to place the plant in cold shutdown is not applicable after thermal power is reduced below 25% RTP. The revised Action requires plant power to be reduced to < 25% RTP (outside the applicable condition) within 4 hours. The current action allows 36 hours to place the plant in a non-applicable condition. As such, this is an additional restriction on plant operation.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Specific"

- L1 The requirement to initiate action within 15 minutes to restore the limit is relaxed and relocated to the Bases in the form of a discussion that "prompt action" should be taken to restore the parameter to within limits. Immediate action may not always be the conservative method to assure safety. The 2 hour completion time for restoration of the limit allows appropriate actions to be evaluated by the operator and completed in a timely manner.

UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



FEB 24 1995

~~3.5/A.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.5.K Minimum Critical Power Ratio (MCPR)

4.5.K Minimum Critical Power Ratio (MCPR)

4.5.K.2 (Cont'd)

b. T as defined in the CORE OPERATING LIMITS REPORT following the conclusion of each scram-time surveillance test required by Specifications 4.3.C.1 and 4.3.C.2.

The determination of the limit must be completed within 72 hours of each scram-time surveillance required by Specification 4.3.C.

See Justification for Change for BFN ISTS 3.2.2

L. APRM Setpoints

1. Applicability { Whenever the core thermal power is $\geq 25\%$ of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0 , or the APRM scram setpoint equation listed in Section 2.1.A and the APRM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT shall be multiplied by FRP/CMFLPD.

LCO 3.2.4

(A2)

2. ACTION A { When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.

3. ACTION B { If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to $\leq 25\%$ of rated thermal power within 4 hours.

(A3)

L. APRM Setpoints

SR 3.2.4.1

FRP/CMFLPD shall be determined daily when the reactor is $\geq 25\%$ of rated thermal power.

(M1) Add proposal 1st Frequency of SR 3.2.4.1

(M2) Proposal SR 3.2.4.2

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



FEB 24 1995

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 Core and Containment Cooling Systems~~

~~4.5 Core and Containment Cooling Systems~~

L. APRM Setpoints

L. APRM Setpoints

Applicability: 1. Whenever the core thermal power is $\geq 25\%$ of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0 , or the APRM scram setpoint equation listed in Section 2.1.A and the APRM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT shall be multiplied by FRP/CMFLPD.

SR 3.2.4.1
FRP/CMFLPD shall be determined daily when the reactor is $\geq 25\%$ of rated thermal power.

LCO 3.2.4

(A2)

(M1) Add proposed 1st Frequency of SR 3.2.4.1

(M2) Proposed SR 3.2.4.2

ACTION A { 2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.

ACTION B { 3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to $\leq 25\%$ of rated thermal power within 4 hours. (A3)

M. Core Thermal-Hydraulic Stability

M. Core Thermal-Hydraulic Stability

1. The reactor shall not be operated at a thermal power and core flow inside of Regions I and II of Figure 3.5.M-1.
2. If Region I of Figure 3.5.M-1 is entered, immediately initiate a manual scram.
3. If Region II of Figure 3.5.M-1 is entered:

1. Verify that the reactor is outside of Region I and II of Figure 3.5.M-1:
 - a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and
 - b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

See Justification for Changes for BFN ISTS 3.4.1



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

FEB 24 1995

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 Core and Containment Cooling Systems~~

L. APERM Setpoints

- 1. Whenever the core thermal power is $\geq 25\%$ of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0 , or the APERM scram setpoint equation listed in Section 2.1.A and the APERM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT shall be multiplied by FRP/CMFLPD.
 - Applicability
 - LCO 3.2.4
 - (A2)
- 2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.
 - Action A
- 3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to $\leq 25\%$ of rated thermal power within 4 hours.
 - Action B
 - (A3)

~~4.5 Core and Containment Cooling Systems~~

L. APERM Setpoints

SR 3.2.4.1
FRP/CMFLPD shall be determined daily when the reactor is $\geq 25\%$ of rated thermal power.

(M1) Add proposed 1st frequency of SR 3.2.4.1

(M2) Proposed SR 3.2.4.2



JUSTIFICATION FOR CHANGES
BFN ISTS 3.2.4 - APRM GAIN AND SETPOINTS

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specification, NUREG 1433. These changes should make the BFN Technical Specifications easier for the operator (and other users) to read and understand. During the reformatting and renumbering process, no technical changes (either actual or interpretational) were made unless they were identified and justified.
- A2 The current LCO and the proposed ISTS LCO ensure acceptable operating margins by limiting excess power peaking or reducing the APRM flow biased neutron flux upscale scram setpoints by the ratio of the fraction of rated power and the core limiting value of the MFLPD. Proposed ISTS LCO Item c also provides the option of increasing the APRM gains to cause the APRM to read ≥ 100 times MFLPD (in %). This condition is to account for the reduction in margin to the fuel cladding integrity safety limit and the fuel cladding 1% plastic strain limit. Either a gain adjustment on the APRMs or an adjustment to the APRM setpoints has effectively the same result. Although BFN Technical Specifications do not specifically call out APRM gain adjustments, they are interpreted as an acceptable alternative and are allowed by current BFN plant procedures. For compliance with proposed LCO Item b (APRM setpoint adjustment) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per proposed LCO 3.3.1.1 (RPS Instrumentation) are required to be adjusted.
- A3 The CTS requirement (CTS 3.5.L.3) to reduce power to $\leq 25\%$ of rated thermal power within 4 hours has been changed $< 25\%$ of rated thermal power consistent with the LCO applicability for the CTS and the proposed BFN ISTS. The intent of the CTS action is to exit the LCO applicability and obviously this cannot be done until power is reduced below 25%. The change is slightly more restrictive by the literal wording of the technical specifications, however, since it does not represent an actual change to the intent it has been classified as an administrative change.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 A new Frequency has been added to require verification of MFLPD within 12 hours of reaching or exceeding 25% RTP. This is an additional restriction on plant operation. CTS would allow up to 24 hours after reaching 25% RTP to perform the test.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.2.4 - APRM GAIN AND SETPOINTS

M2 A new Surveillance Requirement (SR 3.2.4.2) has been added that specifically requires the licensee to verify that APRM setpoint or gains are adjusted for the calculated MFLPD when the method of complying with the LCO is to make these adjustments. Since this change adds a specific requirement where none existed before, the change is considered more restrictive.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.2, POWER DISTRIBUTION LIMITS BASES

The Bases for the current Technical Specifications for this section (3.5.I, 3.5.J, 3.5.K, and 3.5.L) have been completely replaced by revised Bases that reflect the format and applicable content of proposed BFN Unit 2 Technical Specifications Section 3.2, consistent with NUREG-1433. The revised Bases are as shown in the proposed BFN Unit 2 Bases.

50-259
7/6/96

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BROWNS FERRY NUCLEAR PLANT

IMPROVED STANDARD TECHNICAL SPECIFICATIONS

Enclosure V
Volume 18

TENNESSEE VALLEY AUTHORITY

3.6/4.6 PRIMARY SYSTEM BOUNDARY

NOV 18 1988

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

See Justification for Changes for BFN ISTS 3.4.2 and 3.4.3

3.6.E. Jet Pumps

- 1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.D. Relief Valves

- 3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.
- 4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

- 1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

SR 3.4.1.1

(AI)

mismatch Verify

Proposed Note for SR 3.4.1.1

The two recirculation loops have a flow 10% imbalance of 15% or less more when the pumps are operated at the same speed.

(L1)

(M2)

both recirculation loops

- b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.
- c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

AUG 04 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

See Justification for Changes to BFN 1STS 3.4.2

(L1)

Proposed Note to SR 3.4.1.1

4.6.E. Jet Pumps

2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

~~3.6.F Recirculation Pump Operation~~
LCO 3.4.1 ~~matched flow requirement~~

(A1)

ACTIONS C+D

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.

(LA1)

2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

ACTION D

3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation

See Justification for Changes for BFN 1STS 3.4.9

~~4.6.F Recirculation Pump Operation~~

SR 3.4.1.1

(M2)

1. Recirculation pump speeds shall be checked and logged at least once per day.

(A3)

(LA2)

(A1)

~~2. No additional surveillance required.~~

(LA2)

3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.

MAY 31 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.F Recirculation Pump Operation~~

~~3.6.F.3 (Cont'd)~~

temperature of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

ACTION D

see Justification for Changes for BFN ISTS 3.4.9

(m1)

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

ACTION E

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.

a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

See Justification for Changes for CTS 3.6.G/4.6.G in this Section



MAY 31 1994

~~3.5/A.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.M Core Thermal Hydraulic Stability~~

~~4.5.M Core Thermal Hydraulic Stability~~

LCO
3.4.1

1. The reactor shall not be operated at a thermal power and core flow inside of Regions I and II of Figure 3.5.M-1.

Action
A

2. If Region I of Figure 3.5.M-1 is entered, immediately initiate a manual scram.

Action
B

3. If Region II of Figure 3.5.M-1 is entered:

a. Immediately initiate action and exit the region within 2 hours by inserting control rods or by increasing core flow (starting a recirculation pump to exit the region is not an appropriate action), and

LA3

b. While exiting the region, immediately initiate a manual scram if thermal-hydraulic instability is observed, as evidenced by APRM oscillations which exceed 10 percent peak-to-peak of rated or LPRM oscillations which exceed 30 percent peak-to-peak of scale. If periodic LPRM upscale or downscale alarms occur, immediately check the APRM's and individual LPRM's for evidence of thermal-hydraulic instability.

SR 3.4.1.2

1. Verify that the reactor is outside of Region I and II of Figure 3.5.M-1:

a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and

b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

Proposed Action D

L2

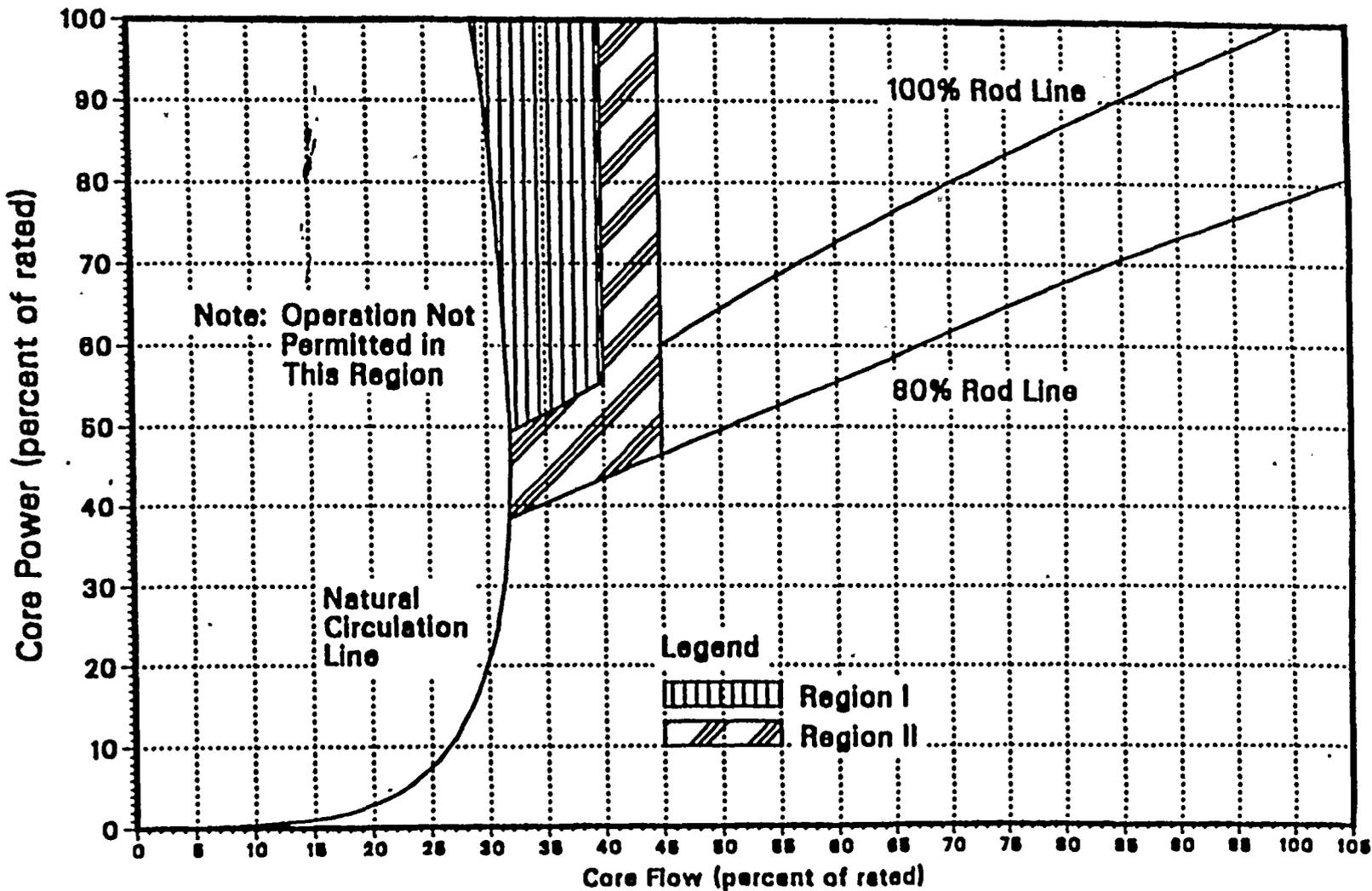
LA3



A1

3.4.1-1

Figure ~~3.5.M-1~~ BFN Power/Flow Stability Regions



BFN
Unit 1

3.5.14.5.22

AMENDMENT NO. 208

Specification 3.4.1

MAY 31 1994

UNIT 2
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

See Justification
for Changes for
BFN ISTS 3.4.2
and 3.4.3

3.6.E. Jet Pumps

1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be shutdown in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.D. Relief Valves

3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.
4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

L1
Proposed Note
for SR 3.4.1.1

SR 3.4.1.1

A1

- a. Verify ^{jet pump} The two recirculation loops have a flow mismatch is imbalance of 15% or less ^{are in} more when the pumps are operated at the same speed. ^{10%} ^{M2}
- b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%. ^{both recirculation loops}
- c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.



AUG 04 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

See Justification for Changes to BFN ISTS 3.4.2

4.6.E. Jet Pumps

2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

(L1) Proposed Note to SR 3.4.1.1

~~3.6.F Recirculation Pump Operation~~

LCO 3.4.1

ACTIONS C+D

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.

2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval, restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature of the reactor

(A1)

~~4.6.F. Recirculation Pump Operation~~

SR 3.4.1.1

1. Recirculation pump speeds shall be checked and logged at least once per day.

(A2)

(M2)

(A3)

(LA2)

2. No additional surveillance required.

(A1)

3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.

(LA2)

See Justification for Changes for BFN ISTS 3.4.9



MAR 18 1993

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(AI)

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.F Recirculation Pump Operation~~

~~3.6.F.3 (Cont'd)~~

vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

ACTION D

(MI)

(See Justification for Changes for BFN 1STS 3.4.9)

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

ACTION E

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a COLD SHUTDOWN CONDITION or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 50.55, except where specific written relief has been granted by 10 CFR 50, Section 50.55a(g)(6)(i).
2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

(See Justification for Changes for CTS 3.6.G/4.6.G in this section)

3.6/4.6-13



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 Core and Containment Cooling Systems~~

~~4.5 Core and Containment Cooling Systems~~

L. APRM Setpoints

1. Whenever the core thermal power is $\geq 25\%$ of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0 , or the APRM scram setpoint equation listed in Section 2.1.A and the APRM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT shall be multiplied by FRP/CMFLPD.
2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.
3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to $\leq 25\%$ of rated thermal power within 4 hours.

L. APRM Setpoints

FRP/CMFLPD shall be determined daily when the reactor is $\geq 25\%$ of rated thermal power.

See Justification for Changes for BFN ISTS 3.2.4

~~M. Core Thermal Hydraulic Stability~~

LCO
3.4.1

1. The reactor shall not be operated at a thermal power and core flow inside of Regions I and II of Figure 3.5.M-1.
2. If Region I of Figure 3.5.M-1 is entered, immediately initiate a manual scram.
3. If Region II of Figure 3.5.M-1 is entered:

ACTION
A

ACTION
B

(A1)

~~M. Core Thermal Hydraulic Stability~~
SR 3.4.1.2

1. Verify that the reactor is outside of Region I and II of Figure 3.5.M-1:
 - a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and
 - b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

OCT 05 1989

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 Core and Containment Cooling Systems~~

~~4.5 Core and Containment Cooling Systems~~

3.5.M.3. (Cont'd)

ACTION B

- a. Immediately initiate action and exit the region within 2 hours by inserting control rods or by increasing core flow (starting a recirculation pump to exit the region is not an appropriate action), and
- b. While exiting the region, immediately initiate a manual scram if thermal-hydraulic instability is observed, as evidenced by APRM oscillations which exceed 10 percent peak-to-peak of rated or LPRM oscillations which exceed 30 percent peak-to-peak of scale. If periodic LPRM upscale or downscale alarms occur, immediately check the APRM's and individual LPRM's for evidence of thermal-hydraulic instability.

Proposed ACTION D

(L2)

LA3

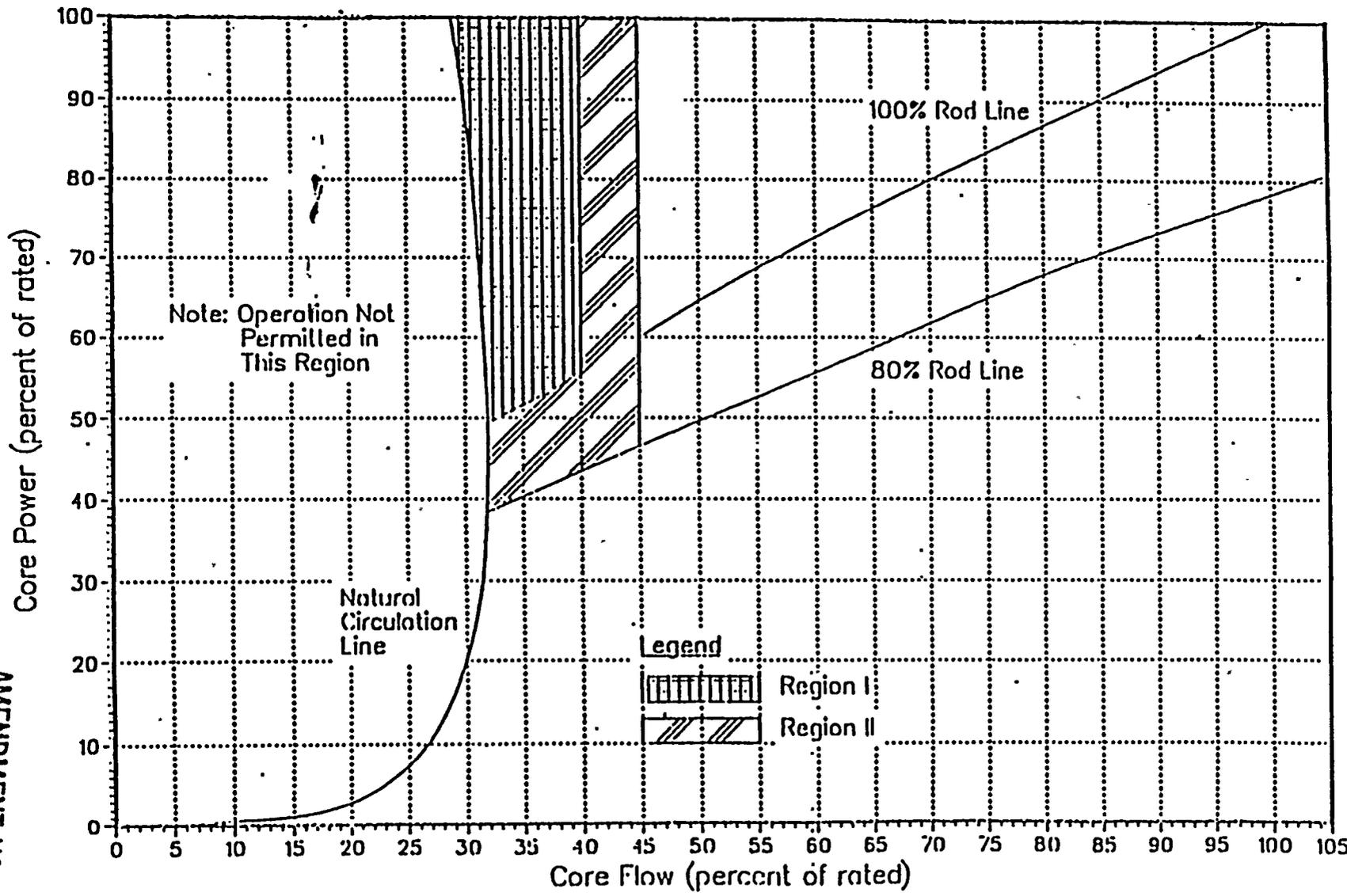
LA3



AI

3.4.1-1

Figure 3.5.M-P
BFN Power/Flow Stability Regions



Legend

Region I

Region II

Note: Operation Not Permitted in This Region

Specification 3.4.1
OCT 05 1989

BFN
Unit 2

3.5/L 5-28

AMENDMENT NO. 174



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

NOV 18 1988

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

See Justification for changes for BFN ISTS 3.4.2 and 3.4.3

3.6.E. Jet Pumps

- 1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is INOPERABLE, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.D. Relief Valves

- 3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.
- 4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

- 1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

M2

SR3.4.1.1

Verify jet pump

Proposed Note for SR3.4.1.1

is 10% mismatch

or 15% or more when the pumps are in operation at the same speed.

L1

both recirculation loops

- a. The two recirculation loops have a flow mismatch of 15% or more when the pumps are in operation at the same speed.

- b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

- c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.



AUG 04 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

See Justification for Changes to BFN ISTS 3.4.2

(L1)

Proposed Note to SR 3.4.1.1

4.6.E. Jet Pumps

2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

(A1)

~~3.6.F Recirculation Pump Operation~~

LCO 3.4.1 (Matched Flow requirements) (M2)

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.

ACTIONS C+D

(A2)

2. Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

(LA1)

3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature

ACTION D

(M1)

See Justification for Changes for BFN ISTS 3.4.9 3.6/4.6-12

~~4.6.F Recirculation Pump Operation~~

SR 3.4.1.1

1. Recirculation pump speeds shall be checked and logged at least once per day.

(M2)

(A3)

(LA2)

(A1)

~~2. No additional surveillance required.~~

(LA2)

3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~
~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.F Recirculation Pump Operation~~

~~3.6.F.3 (Cont'd)~~

of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

ACTION D

See Justification for changes for BFN ISTS 3.4.9

(m1)

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

ACTION E

3.6.G Structural Integrity

- 1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

4.6.G Structural Integrity

- 1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
- 2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

See Justification for changes for CTS 3.6.G/4.6.G in this section



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

MAY 31 1994

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

~~3.5.M Core Thermal-Hydraulic Stability~~

~~4.5.M Core Thermal Hydraulic Stability~~

LCO
3.4.1

Action
A

Action
B

1. The reactor shall not be operated at a thermal power and core flow inside of Regions I and II of Figure 3.5.M-1.

2. If Region I of Figure 3.5.M-1 is entered, immediately initiate a manual scram.

3. If Region II of Figure 3.5.M-1 is entered:

a. Immediately initiate action and exit the region within 2 hours by inserting control rods or by increasing core flow (starting a recirculation pump to exit the region is not an appropriate action), and

b. While exiting the region, immediately initiate a manual scram if thermal-hydraulic instability is observed, as evidenced by APRM oscillations which exceed 10 percent peak-to-peak of rated or LPRM oscillations which exceed 30 percent peak-to-peak of scale. If periodic LPRM upscale or downscale alarms occur, immediately check the APRM's and individual LPRM's for evidence of thermal-hydraulic instability.

SR 3.4.1.2

1. Verify that the reactor is outside of Region I and II of Figure 3.5.M-1:

a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and

b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

(L2)
Proposed Action D

(LA3)

(LA3)

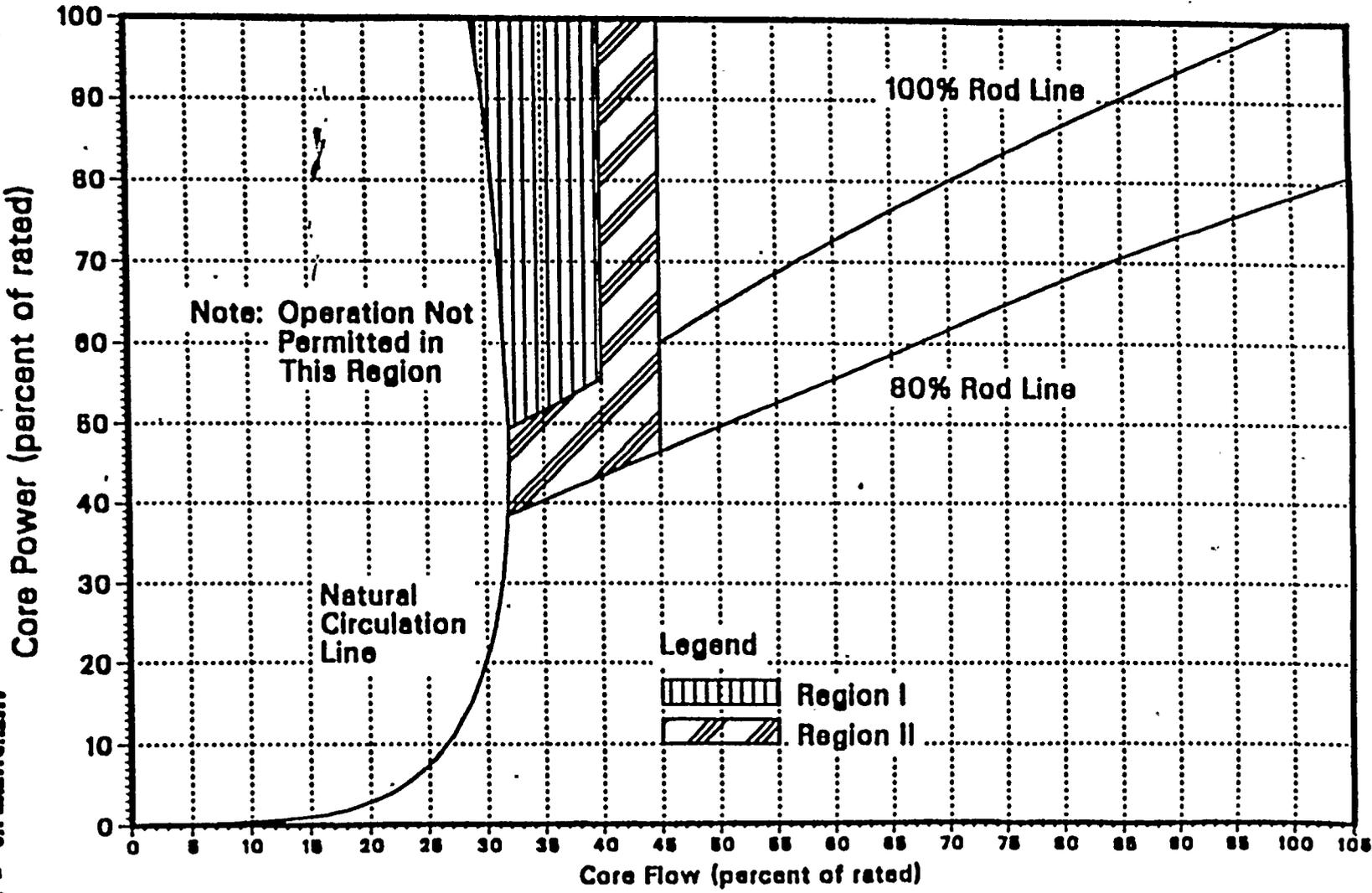


(A1)

3.4.1-1

~~Figure 3.5.M-1~~

BFN Power/Flow Stability Regions



BFN
Unit 3

3.5/4.5-21

AMENDMENT NO. 179

Specification 3.4.1

MAY 31 1994



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.1 - RECIRCULATION LOOPS OPERATING**

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 CTS requires the plant to be placed in the HOT SHUTDOWN CONDITION in 24 hours with one recirculation loop out of service. Proposed ACTION C requires the loop be returned to service in 12 hours or ACTION D requires the plant to be in MODE 3 (Hot Shutdown) in 12 hours. The CTS and the proposed ISTS Completion Times are essentially equivalent since both require the plant to be in MODE 3 in 24 hours.
- A3 The frequency for this Surveillance has been changed from once per day to once per 24 hours. This is a terminology change and is therefore administrative.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 CTS allows up to 24 hours operation with the reactor power < 1% with no recirculation loops operating (the total elapsed time in natural - circulation and one pump operation must be no greater than 24 hours). Proposed ACTION D is more restrictive since the time limit of 12 hours applies to < 1% while in MODE 2 also.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.1 - RECIRCULATION LOOPS OPERATING

- M2 The flow imbalance limit is being reduced to 10% of rated core flow when operating at < 70% of rated core flow, and to 5% of rated core flow when operating at \geq 70% of rated core flow. The current requirement is 15% mismatch of flow at the given flow conditions. While the limit appears to be less restrictive if core flow is \leq 66% of rated core flow, it is more restrictive when $>$ 66% of rated core flow (i.e., 15% x 66% or less is \leq 10% of rated core flow), where the unit normally operates. In addition, currently, this is only a problem if there is an imbalance in combination with two other conditions (CTS 4.6.B.1.b and c). The new requirement is separate from the other two, thus, actions will now be required if there is an imbalance by itself. Therefore, this change is considered more restrictive on plant operations.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

- LA1 This requirement is being relocated to plant specific procedures. The purpose of this limitation is to provide assurance that when shifting from one to two loop operations, excessive vibration of the jet pump risers will not occur. Short term excessive vibration should not result in immediate inoperability of a jet pump, but could reduce the lifetime of the jet pump. This type of requirement is generally found in plant operating procedures, similar to other operating requirements necessary to minimize the potential of damage to components. Changes to the procedures will be controlled by the licensee controlled programs.
- LA2 This requirement is being relocated to plant specific procedures. Details of the methods for performing this Surveillance, and any requirement to record data, has been relocated to plant procedures. Any changes to the procedures will be controlled by the licensee controlled programs.
- LA3 These requirements are being relocated to plant specific procedures. The details of the acceptable method for meeting an action requirement and what constitutes evidence of thermal hydraulic instability and the need to check for it have been relocated to plant procedures. Any changes to the procedures will be controlled by the licensee controlled programs.



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.1 - RECIRCULATION LOOPS OPERATING**

"Specific"

- L1 This change adds a note which states the Surveillance is not required to be performed until 24 hours after both recirculation loops are in operation. The Surveillance is not required to be performed until both loops are in operation since the mismatch limits are meaningless during single loop or natural circulation operation. Also, the Surveillance is allowed to be delayed 24 hours after both recirculation loops are in operation. This allows time to establish appropriate conditions for the test to be performed.
- L2 Per CTS 3.5.M.3.a, if Region II of Figure 3.5.M-1 is not exited within 2 hours, the Specification is violated and CTS 1.0.C.1 applies requiring the plant be placed in Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours. This provides actions for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. The BFN ISTS provides Action within the Specification which could be considered less restrictive than CTS. Action D allows 12 hours to be in MODE 3 (Hot Shutdown) and 36 hours to be in MODE 4 (Cold Shutdown). The proposed Action is considered less restrictive since 12 hours is allowed to place the unit in Hot Shutdown versus the 6 hours allowed to place the unit in Hot Standby per CTS.

UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

NOV 18 1988

~~3.6/A.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

See justification for changes for BFN ISTS 3.4.3

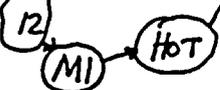
4.6.D. Relief Valves

- The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.
- At least one relief valve shall be disassembled and inspected each operating cycle.

(A1)

~~3.6.E Jet Pumps~~
LCO 3.4.2

Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.



(A1) ~~E. Jet Pumps~~ Verify at least one of the following criteria is satisfied for each operating recirculation loop.

Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

See justification for changes for BFN ISTS 3.4.1

(L2) Proposed SR 3.4.2.1 Notes

(M2) Proposed SR 3.4.2.1 a & c

SR 3.4.2.1 b, d. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

(A3) the established pattern

(L3) less (A2)



AUG 04 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.6.E Jet Pumps~~

(A4)

SR 3.4.2.1

b. 2.

Applicability

Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the

diffuser to lower plenum differential pressure shall be checked

(A5)

daily and the differential pressure of an individual jet pump in a loop shall

(A2)

not vary from the mean of all jet pump

(A3)

differential pressures in that loop by more

than $\frac{10\%}{30}$ (L3)

less (A2)

See Justification for Changes for BFN ISTS 3.4.1

3.6.F Recirculation Pump Operation

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.
2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.
3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation

4.6.F Recirculation Pump Operation

1. Recirculation pump speeds shall be checked and logged at least once per day.
2. No additional surveillance required.
3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



LIMITING CONDITIONS FOR OPERATION

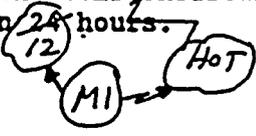
SURVEILLANCE REQUIREMENTS

(A1)

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.4.3

(A1)
3.6.E. Jet Pumps
LCO 3.4.2
+1
Applicability

Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours an orderly shutdown shall be initiated and the reactor shall be shutdown in the GOLD SHUTDOWN CONDITION within 24 hours.



SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.4.1

(L2) Proposed SR 3.4.2.1 Notes

(M2) Proposed SR 3.4.2.1 a + c

4.6.D. Relief Valves

- 3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.
- 4. At least one relief valve shall be disassembled and inspected each operating cycle.

(A1) E. Jet Pumps
Verify at least one of the following criteria is satisfied for each operating recirculation loop

Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

(A2)

a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

(M2)

SR 3.4.2.1

b/c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

(A3) the established pattern

(A2) less

(L3) 20

AUG 04 1994

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

~~4.6.E. Jet Pumps~~

~~SR 3.4.2.1~~
~~b2.~~

Applicability

Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the

diffuser to lower plenum differential pressure shall be checked

daily and the differential pressure of an individual jet pump in a loop shall

not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

(A5)

(A2)

(A3)

(A4)

(A2) (L3) (AZ)

See Justification for Changes for BFN ISTS 3.4.1

3.6.F Recirculation Pump Operation

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.
2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.
3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval, restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature of the reactor

4.6.F. Recirculation Pump Operation

1. Recirculation pump speeds shall be checked and logged at least once per day.
2. No additional surveillance required.
3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



NOV 18 1988

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

See Justification for changes for BFN ISTS 3.4.3

4.6.D. Relief Valves

- The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.
- At least one relief valve shall be disassembled and inspected each operating cycle.

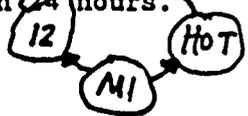
(A1)

~~3.6.E. Jet Pumps~~

LC0 3.4.2

Applicability

Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is INOPERABLE, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.



See Justification for changes For BFN ISTS 3.4.1

(L2) Proposed SR 3.4.2.1 Notes

(M2) Proposed SR 3.4.2.1 a + c

(A1)

~~E. Jet Pumps~~

Verify at least one of the following criteria is satisfied for each operating recirculation loop

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

(M2)

b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

SR 3.4.2.1

b. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than ~~±0% - 20%~~ less

(A3)

the established pattern

(A1)

~~4.6.2. Jet Pumps~~
~~SR 3.4.2.1~~

b-2.

Applicability →

Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than ~~10%~~ 20%.

(A4)

(A5)

(A2)

(A3)

(L3)

(L55)

(A2)

See Justification for changes for BFN 15 TS 3.4.1

3.6.F Recirculation Pump Operation

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.
2. Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.
3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature

4.6.F Recirculation Pump Operation

1. Recirculation pump speeds shall be checked and logged at least once per day.
2. No additional surveillance required.
3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.2 - JET PUMPS**

ADMINISTRATIVE

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change..

- A2 The wording of the surveillance was changed to require verification that one of the following criteria are met rather than verifying that none of the conditions exist simultaneously. This is consistent with NUREG-1433 which attempts to phrase everything in a positive manner. Due to the change in phrasing of the Surveillance, "more than" was changed to "less than or equal to" in criteria b and c.
- A3 The variance of the diffuser-to-lower plenum differential pressure reading on an individual jet pump will now be taken from the established pattern rather than from the mean of all jet pump differential pressures. This change is in accordance with the recommendations of SIL-330 and NUREG/CR-3052 and is consistent with NUREG-1433.
- A4 The conditions of the Surveillance Requirement are assured by LCO 3.4.1. Therefore, there is no need to restate the conditions for jet pump operability.
- A5 The frequency for this Surveillance has been changed from daily to once per 24 hours. This is a terminology change and is therefore administrative.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.2 - JET PUMPS

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 The requirement to place the plant in a Cold Shutdown condition within 24 hours when a jet pump is inoperable has been revised to reflect placing the plant in a non-applicable condition. Current Specification 1.0.C.1 states action requirements are applicable during the operational conditions of each specification. Therefore, the requirement to place the plant in Cold Shutdown is not applicable after Mode 3 is reached. The revised action requires plant power to be brought to Mode 3 (outside the applicable condition) within 12 hours. The current action allows 24 hours to place the plant in a non-applicable condition. As such, this is an additional restriction on plant operation which constitutes a more restrictive change.
- M2 This change adds two requirements to the Surveillance to detect significant degradation in jet pump performance that precedes jet pump failure. The first requirement added would detect a change in the relationship between pump speed, and pump flow and loop flow (difference > 5%). A change in the relationship indicates a plug flow restriction, loss in pump hydraulic performance, leakage, or new flow path between the recirculation pump discharge and jet pump nozzle. The second requirement added monitors the jet pump flow versus established patterns. Any deviations > 10% from normal are considered indicative of potential problem in the recirculation drive flow or jet pump system. These two added requirements to the Surveillance help to detect significant degradation in jet pump performance that precedes jet pump failure. Requirements added to Surveillance Requirements constitute a more restrictive change. In addition, CTS 4.6.E.1 allows jet pump operability to be verified by demonstrating that the two recirculation loops have a flow imbalance of $\leq 15\%$ when the pumps are operated at the same speed. This is now a separate requirement (Proposed SR 3.4.1.1 - See M2 of the Justification for Changes for Specification 3.4.1) and can no longer be used by itself to demonstrate jet pump operability. This change is consistent with NUREG-1433.

SIL-330 provides two alternate testing criteria (thus the deletion of current Surveillance 4.6.E.1.b). One method uses easy to perform surveillances with strict limits to initially screen jet pump operability (the proposed changes above). If these limits are not met, another set of Surveillances exist (current Technical Specifications). Revising the Surveillances to separate the flow imbalance test requirement and to include the stricter limits reflects a more restrictive change.



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.2 - JET PUMPS**

TECHNICAL CHANGE - LESS RESTRICTIVE

"Specific"

L1 This change deletes the current shutdown requirement associated with jet pump flow indication. Currently, when required jet pump flow indication is lost, an orderly shutdown must be initiated in 12 hours and the reactor is required to be in Cold Shutdown within the following 24 hours (since Mode 3 is the non-applicable mode, then 24 hours is allowed to reach Mode 3; see discussion of change M1 for ITS 3.4.2). The proposed Specification implicitly requires the jet pump flow indication to be operable only for the performance of the Surveillance Requirement. If the flow indication is inoperable when the surveillance is required to be performed and jet pump flow can not be determined by other means, the jet pump would be declared inoperable and the appropriate actions would be followed. Since the proposed jet pump surveillance requirement is required to be performed every 24 hours (the 25% extension per SR 3.0.2 can be applied) and the Required Actions require the reactor to be in Mode 3 within 12 hours, the maximum difference in the current Specification and the proposed specification is 6 hours. As a result, the proposed specification effectively allows a maximum of an additional 6 hours (which is the 25% extension) to reach a non-applicable Mode if a required core flow indicator is inoperable and jet pump flow can not be determined. Depending on when the failure occurs, 6 hours is the maximum increase over the current Specifications (failure occurring immediately after the surveillance is performed). The following table provides the details of the calculation of the 6 hour period:

Current Tech Specs	Proposed Tech Specs
Time 0 hours- Jet Pump Indication Fails - 12 hr AOT Begins	Time 0 hours - Jet Pump Indication Fails (Immediately After SR
Time 12 hours- 12 hr AOT Expires - 24 hr AOT Begins to MODE 3 (per 3.0.A; see M1)	Time 30 hours- SR due; Flow Indication Inop (24 hrs x 1.25) - 12 hr AOT to MODE 3 Begins
Time 36 hours- 24 hr AOT Expires Plant in MODE 3	Time 42 hours- 12 hour AOT Expires Plant in MODE 3

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.2 - JET PUMPS**

As depicted above, 42 hours is the maximum time that would be allowed if a required jet pump flow indicator is inoperable and jet pump flow can not be determined. Currently a maximum of 36 hours is allowed if more than one jet pump flow indicator is inoperable. Jet pump flow indication operability does not directly impact jet pump operability. Jet pump flow indication is only required to perform the jet pump Surveillance (SR 3.4.2.1). SR 3.4.2.1 verifies jet pump operability and has a frequency of every 24 hours. The 24 hours frequency plus the 25% extension has been shown by operating experience to be timely for detecting jet pump degradation and is consistent with the surveillance frequency for recirculation loop operability verification. The most common outcome of the performance of a surveillance is the successful demonstration that the acceptance criteria are satisfied. This change is consistent with NUREG-1433.

- L2 Note 1 allows this Surveillance not to be performed until 4 hours after the associated recirculation loop is in operation, since these checks can only be performed during jet pump operation. The 4 hours is an acceptable time to establish conditions appropriate for data collection and evaluation. Note 2 to proposed SR 3.4.2.1 provides time to perform the required Surveillance when the reactor exceeds 25% RTP. Below 25% RTP, low jet pump flow results in indication which precludes the collection of repeatable and meaningful data. The flexibility to proceed to $\geq 25\%$ RTP and then commence the SR every 24 hours is consistent with approved Technical Specifications for both Perry Nuclear Power Plant and River Bend Station.
- L3 The allowed difference between each jet pump diffuser-to-lower plenum differential pressure to the loop average has been increased to 20%. This change is consistent with the recommendations of SIL-330 and NUREG/CR-3052 (Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure). SIL-330 specifies a 10% criteria for individual jet pump flow distribution. When measured by jet pump diffuser-to-lower plenum differential pressure, the equivalent limit is 20% because of the relationship between flow and delta-P. Since BFN uses the diffuser-to-lower plenum differential pressure measurement, the variance allowed should be 20% as recommended by SIL-330 and NUREG/CR-3052. This is a relaxation from existing requirements, therefore, it constitutes a less restrictive change. This increase in allowed difference is considered an acceptable criterion for verifying jet pump operability and is consistent with the BWR Standard Technical Specifications, NUREG 1433.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



DEC 07 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.6.C Coolant Leakage

- 2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

- 3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(A1)

~~3.6.D Relief Valves~~

Action A

- 1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

(L2)

(A1)

Modes 1, 2 + 3

Applicability

Proposed Note to SR 3.4.3.2

(A3)

(A4)

Be in Mode 3 in 12 hrs

(M1)

4.6.C Coolant Leakage

- 2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

See Justification for Changes to BFN 1STS 3.4.4 + 3.4.5

(A1)

(LAI)

~~4.6.D Relief Valves~~
SR 3.4.3.1

- 1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle.

(A2)

All 13 valves will have been checked or replaced upon the completion of every second cycle.

SR 3.4.3.2.2.

- 2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

(A2)

(LAI)



NOV 18 1988

~~(A1) 3.6.4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.6.D. Relief Valves~~

3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.

A5

LA2

4. At least one relief valve shall be disassembled and inspected each operating cycle.

3.6.E. Jet Pumps

1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:
 - a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.
 - b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.
 - c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

See Justification for Changes to BFN ISTS 3.4.2, Jet Pumps



(A1)

1.2/2.2 REACTOR COOLANT SYSTEM INTEGRITY

SAFETY LIMIT

1.2 Reactor Coolant System Integrity

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specifications

- A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

See Justification for Change to BFN ISTS 2.0

LIMITING SAFETY SYSTEM SETTING

2.2 Reactor Coolant System Integrity

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

Limiting Safety Protective Action System Setting
SR 3.4.3.1

- A. Nuclear system relief valves open--nuclear system pressure

1,105 psig ±

33.2 ± psi

(4 valves)

1,115 psig ±

(L1) → 33.5 ± psi

(4 valves)

1,125 psig ±

33.8 ± psi

(5 valves)

- B. Scram--nuclear system high pressure

≤1,055 psig



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



DEC 07 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.6.C Coolant Leakage

- 2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

- 3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.C Coolant Leakage

- 2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

SEE JUSTIFICATION FOR CHANGES TO BFN ISTS 3.4.4 + 3.4.5

(A1) ~~3.6.D Relief Valves~~

- 1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

ACTION A

(L2)

(A1) MODES 1, 2 + 3
Applicability

(A3) Proposed Note to SR 3.4.3.2

(A1) ~~4.6.D Relief Valves~~

SR 3.4.3.1

- 1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

(A2)

SR 3.4.3.2

- 2. In accordance with Specification 1.0.MI, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

(LA1)



LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

~~4.6.9. Relief Valves~~

(A5)

3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.

(LA2)

4. At least one relief valve shall be disassembled and inspected each operating cycle.

3.6.3. Jet Pumps

1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be shutdown in the COLD SHUTDOWN CONDITION within 24 hours.

5. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

- a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.
- b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.
- c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

See Justification for Changes to BFN ISTS 3.4.2, Jet Pumps

1.2/2.2 REACTOR COOLANT SYSTEM INTEGRITY

AI

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

1.2 Reactor Coolant System Integrity

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specifications

- A. The pressure at the lowest point of the reactor vessel shall not exceed 1.375 psig whenever irradiated fuel is in the reactor vessel.

SEE JUSTIFICATION FOR CHANGES TO BFN 1STS 2.0

2.2 Reactor Coolant System Integrity

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

Limiting Safety Protective Action System Setting

SR 3.4.3.1

- A. Nuclear system relief valves $1,105 \text{ psig} \pm 33.2 \text{ psi}$ open--nuclear system pressure (4 valves)

L1 $1,115 \text{ psig} \pm 33.5 \text{ psi}$ (4 valves)

$1,125 \text{ psig} \pm 33.8 \text{ psi}$ (5 valves)

- B. Scram- nuclear system high pressure $\leq 1,055 \text{ psig}$

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.C Coolant Leakage

- 2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

- 3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.C Coolant Leakage

- 2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

See Justification for changes to BFN 1STS 3.4.4 + 3.4.5

(A1)

~~3.6.D. Relief Valves~~

(A4)

be in mode 3 in 12hrs

Action A

- 1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

(L2)

(A1)

modes 1, 2 + 3

Applicability

Proposed Note to SR 3.4.3.2

(A3)

(M1)

A1

~~4.6.D. Relief Valves~~

SR 3.4.3.1

- 1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve, each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

(A2)

SR 3.4.3.2

- 2. In accordance with Specification 1.0.4.1, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve

(LA2)

(LA1)



NOV 18 1988

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

A1

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~A.6.D. Relief Valves~~

A5 3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.

LA2 4. At least one relief valve shall be disassembled and inspected each operating cycle.

3.6.E. Jet Pumps

1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is INOPERABLE, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

See Justification for Changes to BFN ISTRS 3.4.2 Jetpumps

(A1)

Specification 3.4.3

~~1.2/2.2 REACTOR COOLANT SYSTEM INTEGRITY~~

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

1.2 Reactor Coolant System Integrity

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification

- A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

2.2 Reactor Coolant System Integrity

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

SR 3.4.3.1

- A. Nuclear system relief valves $1,105 \text{ psig} \pm 33.2$ psi open—nuclear (4 valves) system pressure

④ $1,115 \text{ psig} \pm 33.5$ psi (4 valves)

$1,125 \text{ psig} \pm 33.3$ psi (5 valves)

- B. Scram—nuclear $\leq 1,055 \text{ psig}$ system high pressure

See justification for charges to BFN ISTS 2.0

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.3 - SAFETY/RELIEF VALVES**

ADMINISTRATIVE

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The Frequency for proposed SR 3.4.3.1 (CTS 4.6.D.1) has been changed from "each operating cycle" to "18 months." Since an operating cycle is 18 months these are equivalent. The Frequency for proposed SR 3.4.3.2 (CTS 4.6.D.2) has been changed from "In accordance with Specification 1.0 MM" to "18 months." Since the Inservice Testing Program (1.0MM) frequency is 18 months these are equivalent. As such, these changes are considered administrative.
- A3 The proposed change adds a note that states that the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME code requirements, prior to valve installation. As such, the addition of the note is considered administrative.
- A4 CTS 3.6.D.1 requires an orderly shutdown when more than one relief valve is known to have failed. Therefore, the CTS allows unlimited operation with one S/RV inoperable. BFN has 13 S/RVs, therefore, 12 are required OPERABLE at all times. LCO 3.4.3 requires 12 to be OPERABLE and shutdown if one of the 12 required S/RVs is inoperable. As such, the two Specifications are equivalent and this change in presentation is considered administrative.

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.3 - SAFETY/RELIEF VALVES**

- A5 BFN CTS 4.6.D.3 is only applicable to three stage Target Rock S/RVs. Only the two stage Target Rock S/RVs are installed and authorized for use in BFN Unit 2. The three stage design is obsolete and is no longer supported at BFN. Since this Surveillance Requirement is no longer applicable to the BFN S/RV design, the deletion of this requirement is considered administrative.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 An additional requirement is being added that requires the plant to be in MODE 3 within 12 hours. This change is more restrictive because it stipulates that the reactor shutdown be completed much earlier than would be required by the existing specifications (CTS 3.6.D.1). CTS requires a shutdown to MODE 4 within 24 hours but does not stipulate how quickly MODE 3 must be reached. Reference Comment L2 which addresses the less restrictive change of be in MODE 4 in 36 hours rather than 24 hours.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

- LA1 The details relating to methods of performing Surveillances have been relocated to the Bases or procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the procedures will be controlled by the licensee controlled programs.
- LA2 This Surveillance Requirement has been relocated to plant procedures since the requirement does not directly relate to S/RV operability. This is strictly a preventive maintenance requirement.

PAGE 2 OF 3

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.3 - SAFETY/RELIEF VALVES

"Specific"

- L1 The allowed lift setpoint tolerance has been increased from 1% to 3% based on incorporation of this larger setpoint tolerance in the BFN reload licensing analysis for each Unit prior to ISTS implementation. The larger setpoint tolerance has already been incorporated into the Unit 2 reload analysis and will be incorporated into the Unit 3 reload analysis for the next cycle (Spring 1997). In addition, when the setpoints are verified, they are still required to be reset to 1% (proposed SR 3.4.3.1). Thus, since the analysis still ensure that all limits are maintained even with the expanded tolerance, this change is considered acceptable. This change is also consistent with the BWR Standard Technical Specifications, NUREG 1433.
- L2 The time to reach MODE 4 (reactor depressurized to < 105 psig, Cold Shutdown) has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added (Reference Comment M4 above). These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.

UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



AUG 26 1987

~~3.0/4.0 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

~~3.6.C. Coolant Leakage~~

Modes 1, 2 + 3

(A1)

~~4.6.G. Coolant Leakage~~

SR 3.4.4.1

1. a. **Applicability** Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

LCO 3.4.4.b

LCO 3.4.4.c

(M3)

(3D) (L1)

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

(LAI)

(L2) (12)

b. Anytime the reactor is in RUN MODE, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN MODE except as defined in 3.6.C.1.c below.

LCO 3.4.4.d

(A2)

Within the previous

c. During the first 24 hours in the RUN MODE following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

LCO 3.4.4.d

(A2)

(M1) Add LCO 3.4.4.a

DEC 07 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1) ~~3.6.C Coolant Leakage~~

(A1) ~~4.6.C Coolant Leakage~~

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

See Justification for changes to BFN ISTS 3.4.5

(L3)

Add Action-A + Required Action B.1

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

ACTION C
(1st Condition)

Add Required Action B.2

Add 2nd Condition of Action C

(L5)

(M1)

3.6.D Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

Hot Shutdown Condition in 12 hours and

4.6.D Relief Valves

(M2)

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

See Justification for changes to BFN ISTS 3.4.3 FN This Section

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

AUG 26 1987

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1) ~~3.6.6. Coolant Leakage~~

Modes 1, 2+3

(A1) ~~3.6.6. Coolant Leakage~~

SR 3.4.4.1

1. a. *Applicability.* Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

LCO 3.4.4.b

LCO 3.4.4.c

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

(LAI)

(L2) (12)

b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below.

LCO 3.4.4.d

(AZ) within the previous

c. During the first 24 hours in the RUN mode following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

LCO 3.4.4.d

(A2)

(M1) Add LCO 3.4.4.e

(L1)

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

(A1) ~~3.6.G Coolant Leakage~~

(A1) ~~4.6.G Coolant Leakage~~

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

SEE JUSTIFICATION FOR CHANGES TO BFN ISTS 3.4.5 IN THIS SECTION

(L3) Add ACTION A + Required Action B.1

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

ACTION C (1st Condition)

Add Required Action B.2

Add 2nd Condition of ACTION C

(L5)

(M1)

(L4) 3.6.D Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

HOT SHUTDOWN CONDITION in 12 hours and

4.6.D Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.FM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

SEE JUSTIFICATION FOR CHANGES TO BFN ISTS 3.4.3 IN THIS SECTION

(M2)



UNIT 3
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

AUG 26 1987

~~3.6/A.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

~~3.6.C Coolant Leakage~~

Modes 1, 2+3

Applicability

1. a.

Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

LCO 3.4.4.b

LCO 3.4.4.c

b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below.

LCO 3.4.4.d

(A2)

within the previous

c. During the first 24 hours in the RUN mode following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

(A2)

LCO 3.4.4.d

(M1) Add LCO 3.4.4.a →

(A1)

~~3.6.C Coolant Leakage~~

SR 3.4.4.1

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

(L1)

(L2)

12

(L1)

(30)

~~2.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

~~3.6.C Coolant Leakage~~

(A1)

~~4.6.C Coolant Leakage~~

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

See Justification for Changes to BFN ISTS 3.4.5 In this Section

(L3)

ADD ACTION A + Required Action B.1

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

ACTION C (1st Condition)

(L4)

3.6.D. Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

HOT SHUTDOWN Condition in 12 hours and

(M2)

See Justification for Changes to BFN ISTS 3.4.3 In this Section

(L5)

ADD Required Action B.2

ADD 2nd Condition of Action C (M1)

4.6.D. Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.4 - RCS OPERATIONAL LEAKAGE

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 The total LEAKAGE limit applies at any moment, to the previous 24 hours (not any future or past 24 hour period). This results in a "rolling average" covering "any 24-hour period." Therefore, changing "any" to "the previous" does not change any intent. In addition, the current provision (CTS 3.6.C.1.c), which allows an increase in reactor coolant leakage into the primary containment of >2 gpm during the first 24 hours in the RUN mode following STARTUP as long as unidentified leakage and total leakage limits are not exceeded, is encompassed by proposed LCO 3.4.4.d which allows the same. LCO 3.4.4.d is worded differently (i.e., ≤ 2 gpm increase in unidentified leakage within the previous 24 hour period in MODE 1) but means the same. Since there is no "previous" 24 hour period until being in MODE 1 for 24 hours, this limit does not apply for the first 24 hours. These are editorial changes only and as such are considered administrative.

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A new requirement has been added to preclude pressure boundary LEAKAGE. An applicable ACTION has also been added. This is an additional restriction on plant operation.

PAGE 1 OF 4



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.4 - RCS OPERATIONAL LEAKAGE

M2 CTS 3.6.C.3 requires an orderly shutdown be initiated and the reactor to be in the COLD SHUTDOWN CONDITION within 24 hours when certain conditions can not be met. Proposed Action C will require the plant be in MODE 3 in 12 hours and MODE 4 in 36 hours. The addition of this intermediate step to the COLD SHUTDOWN CONDITION is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.

M3 The proposed applicability of MODES 1, 2 and 3 is more restrictive than CTS 3.6.C.1.a applicability of "Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F." The Startup Mode will now include the mode switch position of "Refuel" when the head bolts are fully tensioned. The change eliminates the potential to interpret certain plant conditions such that no MODE, or a less restrictive MODE, would exist. Currently, CTS 1.0.M allows the plant to be considered in the SHUTDOWN CONDITION and in the Shutdown Mode with the mode switch in the Refuel position (and other positions are allowed while in the Shutdown Mode) as permitted by notes to that definition.

The allowance to place the Mode Switch in other positions has been moved to Section 3.10, Special Operations and Section 3.3.2.1, Control Rod Block Instrumentation. Any technical changes to these allowances will be discussed in the Justification for Changes to these Sections.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LAI Details of the methods for performing this Surveillance are relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the procedures will be controlled by the licensee controlled programs.

PAGE 2 OF 4



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.4 - RCS OPERATIONAL LEAKAGE

"Specific"

- L1 The total LEAKAGE allowed has been increased to 30 gpm. No applicable safety analysis assumes the total LEAKAGE limit. The limit considers RCS inventory makeup and drywell floor drain capacity. The new limit of 30 gpm is well within the capacity of the Control Rod Drive System pump and the RCIC System, and is well below the capacity of one drywell equipment drain or floor drain pump, which is used to pump the water out of the collecting sump. The collecting sumps can also accommodate this small additional leakage rate.
- L2 The Frequency has been changed from 4 hours to 12 hours, consistent with the allowance in Generic Letter 88-01, Supplement 1. The supplement allows the Frequency to be extended to shiftly, not to exceed 12 hours. Browns Ferry Technical Specifications currently define the frequency of shiftly as 12 hours, thus, this Frequency is adjusted to coincide with this.
- L3 CTS do not provide a period of time to reduce leakage prior to initiating an orderly shutdown. Proposed ACTIONS A and B allow 4 hours to reduce LEAKAGE within limits prior to initiating a shutdown. This is reasonable since the total leakage limits are conservatively below the LEAKAGE that would constitute a critical crack size. The 4 hour completion time for ACTION B is reasonable to properly verify the source of unidentified leakage before the reactor must be shutdown without unduly jeopardizing plant safety. The proposed changes are consistent with the BWR/4 Standard Technical Specifications, NUREG 1433.
- L4 The time allowed to shutdown the plant when the required actions are not met has been changed from "in the COLD SHUTDOWN CONDITION within 24 hours" to in MODE 3 (Hot Shutdown) in 12 hours and MODE 4 (Cold Shutdown) within 36 hours. The proposed allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The additional 12 hours allowed to reach Mode 4 is offset by the safety benefit of being subcritical (MODE 3) in a shorter required time.

PAGE 3 OF 4



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.4 - RCS OPERATIONAL LEAKAGE**

- L5 Proposed LCO 3.4.4, RCS Operational Leakage, will add an alternative to existing requirement in Specifications 3.6.C.1 and 3.6.C.3 that a reactor shutdown be initiated if unidentified leakage increases at a rate of more than 2 gpm within a 24 hour period. Under proposed Required Action B.2, unidentified leakage that increases at a rate of more than 2 gpm within a 24 hour period will not require initiation of a reactor shutdown if it can be determined within 4 hours that the source of the unidentified leakage is not service sensitive type 304 and type 316 austenitic stainless steel piping that is subject to high stress or that contains relatively stagnant or intermittent flow fluids. This alternative Required Action is acceptable because the low limit on the rate of increase of unidentified leakage was established as a method for early identification of Intergranular Stress Corrosion Cracking (IGSCC) in Type 304 and Type 316 austenitic stainless steel piping. IGSCC produces tight cracks and the small flow increase limit is capable of providing an early warning of such deterioration. Verification that the source of leakage is not Type 304 and Type 316 austenitic stainless steel eliminates IGSCC as a cause of leak. This significantly reduces concerns about crack instability and the rapid failure in the RCS boundary. Also, the unidentified LEAKAGE limit is still being maintained and will continue to limit the maximum unidentified LEAKAGE allowed. This change is consistent with NUREG-1433.

PAGE 4 OF 4



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



DEC 07 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1) ~~3.6.C Coolant leakage~~

(Modes 1, 2 + 3) (M1)

(A1) ~~4.6.C Coolant Leakage~~

Applicability

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

Required Action B.1

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

LCD 3.4.6

ACTIONS A+B

Proposed Note to Actions A+B

(A4)

(A6)

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

(L4)

30 days

(M2)

ACTIONS C+D

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(A2)

(M6)

the HOT SHUTDOWN CONDITION in 12 hours and

3.6.D Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

4.6.D Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

See Justification for Changes to BFN ISTS 3.4.3 in this section



TABLE 3.2.E
INSTRUMENTATION THAT MONITORS LEAKAGE INTO DRYWELL

System (2)	Setpoints	Action	Remarks
Equipment Drain Flow Integrator Sump Fill Rate Timer Sump Pump Out Rate Timer	N/A >20.1 min. <13.4 min.	(1)	1. Used to determine identifiable reactor coolant leakage. 2. Considered part of sump system.
Floor Drain Flow Integrator Sump Fill Rate Timer Sump Pump Out Rate Timer	N/A >80.4 min. <8.9 min.	(1)	1. Used to determine unidentifiable reactor coolant leakage. 2. Considered part of sump system.
LCD 3.4.5.a LCD 3.4.5.b Drywell Air Sampling	L3 Or Gas and Particulate 3 x Average Background	(3)	

NOTES:

- (1) Whenever a system is required to be operable, there shall be one operable system either automatic or manual, or ³ ACTION A
- (2) ~~An alternate system to determine the leakage flow is a manual system whereby the time between sump pump starts is monitored. The time interval will determine the leakage flow because the volume of the sump will be known.~~ LA1
- (3) Upon receipt of alarm, immediate action will be taken to confirm the alarm and assess the possibility of increased leakage. LA3

BFW Unit 1



TABLE 4.2.E
 MINIMUM TEST AND CALIBRATION FREQUENCY FOR DRYWELL LEAK DETECTION INSTRUMENTATION

A3

SR 3.4.5.1

Function	Functional test SR 3.4.5.3	Calibration SR 3.4.5.4	Instrument Check SR 3.4.5.1
LAS → Equipment Drain Sump Flow Integrator	(1)	once/6 months	once/day
Floor Drain Sump Flow Integrator	(4) LA4	once/6 months (184 days) AS	once/day L2
Air Sampling System SR 3.4.5.2 →	(1) 31 days AS	once/3 months (18 months) LS	once/day
LAS → Equipment Drain Sump Fill Rate and Pumpout Rate Timers	(4)	once/operating cycle	N/A (12 hrs) M5
LA4 → Floor Drain Sump Fill Rate and Pumpout Rate Timers	(4)	once/operating cycle	N/A
LAS → Equipment Drain Logic	once/operating cycle	(6)	N/A
LA4 → Floor Drain Logic	once/operating cycle	(6)	N/A

3.2/4.2-53

BFN Unit 1

FEB 05 1987

Specification 3.4.5

only Notes 1, 4 and 6 apply to Table 4.2.E. The remaining Notes are addressed in the markups to Section 3.3 Instrumentation.

Justification 3.7.3

JAN 26 1989

(A1)

~~NOTES FOR TABLES 4.2.A THROUGH 4.2.L except 4.2.D AND 4.2.E~~
SR 3.4.5.2 Free

31 days

(A5)

1. Functional tests shall be performed once per month.
2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.
3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

~~4. Tested during logic system functional tests.~~

(LA4)

5. Refer to Table 4.1.B.

~~6. The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.~~

7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.
8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be OPERABLE or are tripped.
9. Calibration frequency shall be once/year.
10. Deleted
11. Portion of the logic is functionally tested during outage only.
12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.
13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

See Justification for Changes for BFN ISTS 3.3

BFN Unit 1

3.2/4.2-59

AMENDMENT NO. 164



UNIT 2
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

DEC 07 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

(A1) ~~3.6.6 Coolant Leakage~~

Modes 1, 2+3

(M1)

Applicability

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

LCO 3.4.6

ACTIONS A+B

Proposed Note to ACTION B

(A1)

(A6)

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

ACTIONS C+D

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(L1)

3.6.D Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

SEE JUSTIFICATION FOR CHANGES TO BFN ISTS 2.4.3 IN THIS SECTION

(A1) ~~4.6.6 Coolant Leakage~~

Required Action B.1

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

12

(M2)

30 days

(L4)

(A2)

(M6)

the HOT SHUTDOWN CONDITION in 12 hours and

4.6.D Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.



TABLE 3.2.E
INSTRUMENTATION THAT MONITORS LEAKAGE INTO DRYWELL

System (2)	Setpoints	Action	Remarks
Equipment Drain Flow Integrator Sump Fill Rate Timer Sump Pump Out Rate Timer	N/A ≥20.1 min. ≤13.4 min.	(1)	1. Used to determine identifiable reactor coolant leakage. 2. Considered part of sump system.
Floor Drain Flow Integrator Sump Fill Rate Timer Sump Pump Out Rate Timer	N/A ≥80.4 min. ≤8.9 min.	(1)	1. Used to determine unidentifiable reactor coolant leakage. 2. Considered part of sump system.
Drywell Air Sampling	Gas and Particulate 3 X Average Background	(3)	

LCO 3.4.5.a

LCO 3.4.5.b
3.2/4.2-30

NOTES:

- (1) Whenever a system is required to be operable, there shall be one operable system either automatic or manual, or the action required in Section 3.6.C.2 shall be taken. } ACTION A
- (2) An alternate system to determine the leakage flow is a manual system whereby the time between sump pump starts is monitored. The time interval will determine the leakage flow because the volume of the sump will be known. LA1
- (3) Upon receipt of alarm, immediate action will be taken to confirm the alarm and assess the possibility of increased leakage. LA3

BFN-Unit 2

TABLE 4.2.E
 MINIMUM TEST AND CALIBRATION FREQUENCY FOR DRYWELL LEAK DETECTION INSTRUMENTATION

A3

Function	Functional Test	SR 3.4.5.3 Calibration	SR 3.4.5.4 Instrument Check
LA5 Equipment Drain Sump Flow Integrator	(4)	once/6 months	once/day
Floor Drain Sump Flow Integrator	(4) LA4	once/6 months AS	once/day L2
Air Sampling System SR 3.4.5.2	(4) 31 days AS	once/3 months AS	once/day 12 hrs L5
LA5 Equipment Drain Sump Fill Rate and Pumpout Rate Timers	(4)	once/operating cycle	N/A MS
LA4 Floor Drain Sump Fill Rate and Pumpout Rate Timers	(4)	once/operating cycle	N/A
LA5 Equipment Drain Logic	once/operating cycle	(6)	N/A
LA4 Floor Drain Logic	once/operating cycle	(6)	N/A

3.2/4.2-53

BFN-Unit 2



Only Notes 1, 4 and 6 apply to Table 4.2.E. The remaining Notes are addressed in the manuscripts to Section 3.3, Instrumentation.

Specification 34.5

(A1)

~~NOTES FOR TABLES 4.2.A THROUGH 4.2.I except 4.2.D AND 4.2.K~~

JAN 26 1989

SR 34.5.2 Freq.

1. Functional tests shall be performed once per ~~month~~ ^{31 days} (AS).
2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.
3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.
4. Tested during logic system functional tests. (LAY)
5. Refer to Table 4.1.B.
6. The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.
7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.
8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be OPERABLE or are tripped.
9. Calibration frequency shall be once/year.
10. Deleted
11. Portion of the logic is functionally tested during outage only.
12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.
13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

See Justification for Changes
for BFN 1STS 3.3

UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

~~2.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.C Coolant Leakage~~ (Modes 1, 2, & 3) (m)

~~4.6.C Coolant Leakage~~ (A1)

Applicability 2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

LCD 3.4.6

ACTIONS A+B

Proposed Note to Actions A+B

(A4)

(A6)

(L4)

30 days

Required Action B.1

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

(M2)

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

(A2)

(M6)

ACTIONS C+D

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(L1)

the HBT SHUTDOWN CONDITION in 12 hours and

~~3.6.D. Relief Valves~~

~~4.6.D. Relief Valves~~

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve

See Justification for CHANGES TO BEN 1STS 3.4.3 in this Section



TABLE 3.2.E
INSTRUMENTATION THAT MONITORS LEAKAGE INTO DRYWELL

(A3)

System (2)	Setpoints	Action	Remarks
Equipment Drain Flow Integrator Sump Fill Rate Timer Sump Pump Out Rate Timer	N/A ≥20.1 min. ≤13.4 min.	(1)	1. Used to determine identifiable reactor coolant leakage. 2. Considered part of sump system.
Floor Drain Flow Integrator Sump Fill Rate Timer Sump Pump Out Rate Timer	N/A ≥60.4 min. ≤8.9 min.	(1)	1. Used to determine unidentifiable reactor coolant leakage. 2. Considered part of sump system.
LCO 3.4.5.a LCO 3.4.5.b Drywell Air Sampling	2 of Gas and Particulate 3 x Average Background	(3)	

(LA5)

(LA2)

(A3)

(L3)

(3)

NOTES:

(1) Whenever a system is required to be operable, there shall be one operable system either automatic or manual, or the action required in Section 3.6.C.2 shall be taken.

Action A

(2) An alternate system to determine the leakage flow is a manual system whereby the time between sump pump starts is monitored. The time interval will determine the leakage flow because the volume of the sump will be known.

(LA1)

(3) Upon receipt of alarm, immediate action will be taken to confirm the alarm and assess the possibility of increased leakage.

(LA3)

BFN-Unit 3



TABLE 4.2.E
MINIMUM TEST AND CALIBRATION FREQUENCY FOR DRYWELL LEAK DETECTION INSTRUMENTATION

A3

SR 3.4.5.1

Function	Functional Test SR 3.4.5.3	Calibration SR 3.4.5.4	Instrument Check SR 3.4.5.1
LA5 Equipment Drain Sump Flow Integrator	(6)	once/6 months	once/day
Floor Drain Sump Flow Integrator	(4) LA4	once/6 months AS 187 days	once/day L2
Air Sampling System SR 3.4.5.2	(1) 31 days AS	18 once/3 months LS	once/day 12hrs MS
LA5 Equipment Drain Sump Fill Rate and Pumpout Rate Timers	(4)	once/operating cycle	N/A
LA4 Floor Drain Sump Fill Rate and Pumpout Rate Timers	(4)	once/operating cycle	N/A
LA5 Equipment Drain Logic	once/operating cycle	(6)	N/A
LA4 Floor Drain Logic	once/operating cycle	(6)	N/A

3.2/4.2-52

BFM-Unit 3



Only Notes 1, 4 and 6 apply to Table 4.2.E. The remaining notes are addressed in the markups to Section 3.3, Instrumentation.

Specification 3.4.5

(A1)

~~NOTES FOR TABLES 4.2.A THROUGH 4.2.I except 4.2.D AND 4.2.K~~

JAN 26 1989

SR 3.4.5.2 Free

1. Functional tests shall be performed once per ~~month~~ ^{31 days} (45)

2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. ~~Tested during logic system functional tests.~~

(LA4)

5. Refer to Table 4.1.B.

6. ~~The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.~~

7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.

8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be operable or are tripped.

9. Calibration frequency shall be once/year.

10. (DELETED)

11. Portion of the logic is functionally tested during outage only.

12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.

13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

See Justification for Changes
for BFN ISTS 3.3



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 The revised presentation of actions is proposed to explicitly identify that LCO 3:0.3 is required to be entered if all required RCS leakage monitoring systems are inoperable. This action is consistent with the current requirements and is considered a presentation preference. Therefore, this change is considered administrative.

A3 The Table format is being deleted. This change is considered a presentation preference. Therefore, this change is considered administrative.

A4 Proposed ACTION B is modified by a note that explicitly states that the provisions of 3.0.4 are not applicable. This explicitly allows a mode change when both the particulate and gaseous primary containment monitoring channels are inoperable. This allowance is provided because, in this Condition, the drywell sump monitoring system will be available to monitor RCS leakage and the compensatory actions for the inoperable system will provide additional indication of RCS leakage. This is an administrative change since existing Technical Specifications do not have an explicit requirement that prohibits entry into a Mode or condition when an LCO required by that Mode or condition is not satisfied. Therefore, CTS allows the actions being permitted by the note being added. This is consistent with NUREG-1433.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

- A5 Frequency has been editorially changed from monthly to every 31 days and from every six months to every 184 days. This is an administrative change since these are equivalent time periods.
- A6 The current provision (CTS 3.6.C.2, 2nd paragraph) that allows the air sampling system to be removed from service for a period of 4 hours for calibration, functional testing, and maintenance without providing a temporary monitor has been eliminated. There is currently no requirement for a monitor for at least 24 hours (CTS 4.6.C.2). Therefore, the current provision serves no purpose.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 The proposed applicability of MODES 1, 2 and 3 is more restrictive than CTS 3.6.C.1.a applicability of "Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F." The Startup Mode will now include the mode switch position of "Refuel" when the head bolts are fully tensioned. The change eliminates the potential to interpret certain plant conditions such that no MODE, or a less restrictive MODE, would exist. Currently, CTS 1.0.M allows the plant to be considered in the SHUTDOWN CONDITION and in the Shutdown Mode with the mode switch in the Refuel position (and other positions are allowed while in the Shutdown Mode) as permitted by notes to that definition.

The allowance to place the Mode Switch in other positions has been moved to Section 3.10, Special Operations and Section 3.3.2.1, Control Rod Block Instrumentation. Any technical changes to these allowances will be discussed in the Justification for Changes to these Sections.

- M2 The frequency of grab sampling with the air sampling system inoperable has been increased from 24 hours to 12 hours. A grab sample once/12 hours provides adequate information to detect leakage during the extended (See Justification for Change L4) period of time that the air sampling system is allowed to be inoperable.
- M3 Not used.
- M4 Not used.
- M5 The Frequency of the channel check requirement has been changed from every 24 hours to every 12 hours, consistent with Generic Letter 88-01, Supplement 1 and NUREG-1433. This is an additional restriction on plant operation.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

M6 CTS 3.6.C.3 requires an orderly shutdown be initiated and the reactor to be in the COLD SHUTDOWN CONDITION within 24 hours when certain conditions can not be met. Proposed Action C will require the plant be in MODE 3 in 12 hours and MODE 4 in 36 hours. The addition of this intermediate step to the COLD SHUTDOWN CONDITION is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

- LA1 The description of an acceptable alternate system to measure leakage has been relocated to the Bases or procedures that support compliance with the limits for RCS Operational Leakage in proposed Specification 3.4.4. The design features and system operation are also described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the procedures and FSAR will be controlled by the provisions of 10 CFR 50.59.
- LA2 The details relating to the setpoints have been relocated to the procedures. Changes to the procedures will be controlled by the licensee controlled programs.
- LA3 The details relating to actions required upon receipt of an alarm have been relocated to procedures. Changes to the procedures will be controlled by the licensee controlled programs.
- LA4 Details of the specifics of the functional, calibration, and logic system functional test related to the floor drain sump fill rate and pump out timers has been relocated to procedures since the operability of the system is not dependent upon these timers. Changes to the procedures will be controlled by the licensee controlled programs.



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION**

- LA5 The drywell equipment drain sump monitoring system functions to quantify identified leakage. Since the purpose of this specification is to provide early indication of unidentified RCS leakage, the drywell equipment drain sump monitoring system has been relocated to the Bases or procedures that support compliance with the limits for RCS Operational Leakage in proposed Specification 3.4.4. The design features and system operation are also described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the procedures and FSAR will be controlled by the provisions of 10CFR50.59.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Specific"

- L1 The time allowed to shutdown the plant when the required actions are not met has been changed from "in the COLD SHUTDOWN CONDITION within 24 hours" to in MODE 3 (Hot Shutdown) in 12 hours and MODE 4 (Cold Shutdown) within 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added. These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.
- L2 This requirement has been deleted. An instrument check would not consistently demonstrate operability since normally the instruments could not be compared to any other instruments, and their reading could be anywhere on scale; thus, observing the meter would provide no valid information as to whether the instrument is OPERABLE. The CHANNEL FUNCTIONAL TEST requirement is the best indicator of OPERABILITY while operating, and this requirement is being maintained. This is also consistent with the BWR Standard Technical Specification, NUREG 1433.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

- L3 CTS Table 3.2.E defines the air sampling system as consisting of gas and particulate monitoring channels (i.e., both channels are required OPERABLE for the air sampling system to be considered OPERABLE). Proposed LCO 3.4.5.b requires either one channel of the gas or one channel of the particulate monitoring system to be OPERABLE. This is less restrictive than CTS requirements but is acceptable since either channel is capable of indicating increased LEAKAGE rates that correlate to radioactivity levels of 3 times average background.
- L4 The allowed outage time for the air sampling system has been changed from 72 hours to 30 days. The 30 day allowed outage time recognizes that at least one other form of leak detection is available (sump monitoring) and takes credit for the increased sampling frequency of 12 hours (versus CTS of 24 hrs). This change is consistent with NUREG-1433.
- L5 The calibration frequency has been changed once per 3 months to once per 18 months. This new Frequency is consistent with BFN setpoint methodology, which considers the magnitude of the equipment drift in the setpoint analysis over an 18 month calibration interval. The primary containment leak detection noble gas and particulate monitor is a digital Eberline continuous air monitor (CAM) which is identical to the building effluent monitors whose calibration frequency is 18 months in accordance with the Offsite Dose Calculation Manual (ODCM) and previously required by Technical Specification Table 4.2.K until these instruments were removed by Amendment No. 216 dated September 22, 1993 (reference TS 301). Excessive calibration can cause damage to the equipment. In addition, plant operations could be impacted while the equipment is removed from service for calibration since it would not be available for leak detection.



UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.B. Coolant Chemistry

4. When the reactor is not pressurized with fuel in the reactor vessel, except during the STARTUP CONDITION, the reactor water shall be maintained within the following limits.

- a. Conductivity - 10 μ mho/cm at 25°C
- b. Chloride - 0.5 ppm
- c. pH shall be between 5.3 and 8.6.

5. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated immediately. The reactor shall be brought to the COLD SHUTDOWN CONDITION as rapidly as cooldown rate permits.

LCO
3.4.6
+
Applic

6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 μ Ci/gm of dose equivalent I-131.

(M1)

4.6.B. Coolant Chemistry

4. Whenever the reactor is not pressurized with fuel in the reactor vessel, a sample of the reactor coolant shall be analyzed at least every 96 hours for conductivity, chloride ion content and pH.

See Justification for Changes for CTS 3.6.8/4.6.8 in this Section

Note for SR 3.4.9.1

(M2)

SR 3.4.6.1

5. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least I-131, I-132, I-133, and I-134 shall be performed monthly on a coolant liquid sample.

(LAI)

(M3)

7 days

Required Action A.1+B.1

6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:

(LAI)



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.B. Coolant Chemistry~~ (A2)

~~4.6.B. Coolant Chemistry~~

3.6.B.6 (Cont'd) Proposed Note to Required Actions for Condition A

~~4.6.B.6 (Cont'd)~~

ACTION A

(L1)

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 $\mu\text{Ci/gm}$ whenever the reactor is critical. The reactor shall not be operated more than 5% of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 $\mu\text{Ci/gm}$, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

ACTION B

(L2) within 12 hours

(A3) or be in Mode 4 within 36 hours

(LAI)

(MI)

Rea Act A.1

(LAI)

- a. During the STARTUP CONDITION
- b. Following a significant power change**
- c. Following an increase in the equilibrium off-gas level exceeding 10,000 $\mu\text{Ci/sec}$ (at the steam jet air ejector) within a 48-hour period

d. Whenever the equilibrium iodine limit specified in 3.6.B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 $\mu\text{Ci/gm}$) is established. However, at least 3 consecutive samples shall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration.

7. When there is no fuel in the reactor vessel, technical specification reactor coolant chemistry limits do not apply.

7. When there is no fuel in the reactor vessel, sampling of reactor coolant chemistry at technical specification frequency is not required.

See Justification for Changes for CTS 3.6.B/4.6.B in this Section

** For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.

(LAI)



(M1)

INSERT PROPOSED NEW SPECIFICATION 3.4.7

Insert new Specification 3.4.7, Residual Heat Removal System - Hot Shutdown, as shown in the BFN Unit 2 Improved Technical Specifications.

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.7
RHR SHUTDOWN COOLING SYSTEM - HOT SHUTDOWN**

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 A new Specification is being added requiring two RHR Shutdown Cooling subsystems to be OPERABLE in MODE 3 with reactor steam dome pressure less than the RHR low pressure permissive pressure. Appropriate ACTIONS and a Surveillance Requirement are also added. This is consistent with the BWR Standard Technical Specification, NUREG 1433 and is an additional restriction on plant operation.

(MII)

INSERT PROPOSED NEW SPECIFICATION 3.4.8

Insert new Specification 3.4.8, Residual Heat Removal System - Cold Shutdown, as shown in the BFN Unit 2 Improved Technical Specifications.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.8
RHR SHUTDOWN COOLING SYSTEM - COLD SHUTDOWN

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A new Specification is being added requiring two RHR Shutdown Cooling subsystems to be OPERABLE in MODE 4. Appropriate ACTIONS and a Surveillance Requirement are also added. This is consistent with the BWR Standard Technical Specification, NUREG 1433 and is an additional restriction on plant operation.



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

JUN 28 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.6.B. Coolant Chemistry

4. When the reactor is not pressurized with fuel in the reactor vessel, except during the STARTUP CONDITION, the reactor water shall be maintained within the following limits.

- a. Conductivity - 10 μ mho/cm at 25°C
- b. Chloride - 0.5 ppm
- c. pH shall be between 5.3 and 8.6.

5. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated immediately. The reactor shall be brought to the COLD SHUTDOWN CONDITION as rapidly as cooldown rate permits.

6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 μ Ci/gm of dose equivalent I-131.

LCO
3.4.6
+
Applic.

(M1)

4.6.B. Coolant Chemistry

4. Whenever the reactor is not pressurized with fuel in the reactor vessel, a sample of the reactor coolant shall be analyzed at least every 96 hours for conductivity, chloride ion content and pH.

See Justification for Changes for CTS 3.6.B/4.6.B in this section

(M2)

Note for SR 3.4.9.1

SR 3.4.6.1

5. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least I-131, I-132, I-133, and I-134 shall be performed monthly on a coolant liquid sample.

(LAI)

(M3)

every 7 days

Required Action A.1 + B.1

6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:

(LAI)



JUN 28 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.B. Coolant Chemistry~~ (A2)

~~4.6.B. Coolant Chemistry~~

~~3.6.B.6 (Cont'd)~~

Proposed Note to Required Action for Condition A

~~4.6.B.6 (Cont'd)~~

ACTION A

L1

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 $\mu\text{Ci/gm}$ whenever the reactor is critical. The reactor shall not be operated more than 5% of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 $\mu\text{Ci/gm}$, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

ACTION B

L2

within 12 hours

A3

or be in MUDS 4 with 36 hours

A1

LAI

MI

Req. Act. A.1

LAI

- a. During the STARTUP CONDITION
- b. Following a significant power change**
- c. Following an increase in the equilibrium off-gas level exceeding 10,000 $\mu\text{Ci/sec}$ (at the steam jet air ejector) within a 48-hour period.
- d. Whenever the equilibrium iodine limit specified in 3.6.B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 $\mu\text{Ci/gm}$) is established. However, at least 3 consecutive samples shall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration.

7. When there is no fuel in the reactor vessel, technical specification reactor coolant chemistry limits do not apply.

7. When there is no fuel in the reactor vessel, sampling of reactor coolant chemistry at technical specification frequency is not required.

See Justification for Changes for CTS 3.6B/4.6.B in this Section

LAI

** For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.



(M1)

INSERT PROPOSED NEW SPECIFICATION 3.4.7

Insert new Specification 3.4.7, Residual Heat Removal System - Hot Shutdown, as shown in the BFN Unit 2 Improved Technical Specifications.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.7
RHR SHUTDOWN COOLING SYSTEM - HOT SHUTDOWN

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A new Specification is being added requiring two RHR Shutdown Cooling subsystems to be OPERABLE in MODE 3 with reactor steam dome pressure less than the RHR low pressure permissive pressure. Appropriate ACTIONS and a Surveillance Requirement are also added. This is consistent with the BWR Standard Technical Specification, NUREG 1433 and is an additional restriction on plant operation.



(M1)

INSERT PROPOSED NEW SPECIFICATION 3.4.8

Insert new Specification 3.4.8, Residual Heat Removal System - Cold Shutdown, as shown in the BFN Unit 2 Improved Technical Specifications.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.8
RHR SHUTDOWN COOLING SYSTEM - COLD SHUTDOWN

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A new Specification is being added requiring two RHR Shutdown Cooling subsystems to be OPERABLE in MODE 4. Appropriate ACTIONS and a Surveillance Requirement are also added. This is consistent with the BWR Standard Technical Specification, NUREG 1433 and is an additional restriction on plant operation.

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



(A1)

3.6.B. Coolant Chemistry

4. When the reactor is not pressurized with fuel in the reactor vessel, except during the STARTUP CONDITION, the reactor water shall be maintained within the following limits.

- a. Conductivity - 10 μ mho/cm at 25°C
- b. Chloride - 0.5 ppm
- c. pH shall be between 5.3 and 8.6.

5. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated immediately. The reactor shall be brought to the COLD SHUTDOWN CONDITION as rapidly as cooldown rate permits.

6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 μ Ci/gm of dose equivalent I-131.

LCO
3.4.6
+
Applicability

(M1)

4.6.B. Coolant Chemistry

4. Whenever the reactor is not pressurized with fuel in the reactor vessel, a sample of the reactor coolant shall be analyzed at least every 96 hours for conductivity, chloride ion content and pH.

See Justification for Changes for CTS 3.6.B/4.6.B in this section

(M2)

Note for SR 3.4.9.1

SR 3.4.6.1

5. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least I-131, I-132, I-133, and I-134 shall be performed monthly on a coolant liquid sample.

(LAI)

(M3) every 7 days

Required Action A.1 & B.1

6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:

(LAI)



3.6.B. Coolant Chemistry

4.6.B. Coolant Chemistry

3.6.B.6 (Cont'd)

4.6.B.6 (Cont'd)

Proposed Note to Required Actions for Condition A

Action A

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 µCi/gm whenever the reactor is critical. The reactor shall not be operated more than 5% of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 µCi/gm, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

L1

LAI

M1

Action B

If the iodine concentration in the coolant exceeds 26 µCi/gm, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

L2 - Within 12 hours

A3 - or be in Mode 4 within 36 hours

Req Action A.1

LAI

- a. During the STARTUP CONDITION
- b. Following a significant power change**
- c. Following an increase in the equilibrium off-gas level exceeding 10,000 µCi/sec (at the steam jet air ejector) within a 48-hour period.
- d. Whenever the equilibrium iodine limit specified in 3.6:B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 µCi/gm) is established. However, at least 3 consecutive samples shall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration.

7. When there is no fuel in the reactor vessel, technical specification reactor coolant chemistry limits do not apply.

7. When there is no fuel in the reactor vessel, sampling of reactor coolant chemistry at technical specification frequency is not required.

See Justification for Changes for CTS 3.6.B/4.6.B in this Section.

** For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.

LAI



(M1)

INSERT PROPOSED NEW SPECIFICATION 3.4.7

Insert new Specification 3.4.7, Residual Heat Removal System - Hot Shutdown, as shown in the BFN Unit 2 Improved Technical Specifications.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.7
RHR SHUTDOWN COOLING SYSTEM - HOT SHUTDOWN

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A new Specification is being added requiring two RHR Shutdown Cooling subsystems to be OPERABLE in MODE 3 with reactor steam dome pressure less than the RHR low pressure permissive pressure. Appropriate ACTIONS and a Surveillance Requirement are also added. This is consistent with the BWR Standard Technical Specification, NUREG 1433 and is an additional restriction on plant operation.



(M11)

INSERT PROPOSED NEW SPECIFICATION 3.4.8

Insert new Specification 3.4.8, Residual Heat Removal System - Cold Shutdown, as shown in the BFN Unit 2 Improved Technical Specifications.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.8
RHR SHUTDOWN COOLING SYSTEM - COLD SHUTDOWN

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A new Specification is being added requiring two RHR Shutdown Cooling subsystems to be OPERABLE in MODE 4. Appropriate ACTIONS and a Surveillance Requirement are also added. This is consistent with the BWR Standard Technical Specification, NUREG 1433 and is an additional restriction on plant operation.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.6 - RCS SPECIFIC ACTIVITY

ADMINISTRATIVE

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 Note is added to the Required Actions for Condition A to indicate that LCO 3.0.4 is not applicable. Entry into the Applicable Modes should not be restricted since the most likely response to the condition is restoration of compliance within the allowed 48 hours. Further, since the LCO limits assure the dose due to a LOCA would be a small fraction of the 10 CFR 100 limit, operation during the allowed time frame would not represent a significant impact to the health and safety of the public. In addition, this allowance is already inherently provided by the words of Specification 4.6.B.6.a, which states that additional samples are required "during startup" when specific activity exceeds the limit. Thus, this change is a presentation preference only and is considered administrative.

- A3 Existing Specification 3.6.B.6 requires that if the Dose Equivalent I-131 cannot be restored within 48 hours, or if at any time it exceeds 26 $\mu\text{Ci/gm}$, the reactor must be shut down and all main steam lines must be isolated immediately. Proposed LCO 3.4.6, Condition B, allows the alternative of being in MODE 3 within 12 hours and Mode 4 within 36 hours under the same conditions. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In Mode 4, the LCO requirements are no longer applicable. This change is considered administrative because existing 1.0.C.1 would require that the reactor be placed in Mode 4 within 36

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.6 - RCS SPECIFIC ACTIVITY**

hours if the requirements in CTS 3.6.B.6 could not be met. This change is consistent with NUREG-1433.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 The Applicability has been changed to require the specific activity to be within limits in those conditions which represent a potential for release of significant quantities of radioactive coolant to the environment. Thus, MODE 3 with any steam line not isolated has been added. In addition, MODE 2 with any steam line not isolated has been added in lieu of MODE 2 when the reactor is critical. While this does allow the reactor to be critical with the main steam lines isolated while not requiring the LCO to be met, overall this change is considered more restrictive due to the MODE 2 subcritical and MODE 3 requirements. In addition, the ACTIONS have been modified to reflect the new Applicability, and an option for exiting the applicable MODES is provided for cases where isolation is not desired.
- M2 CTS 4.6.B.5 requires sampling reactor coolant to determine specific activity "during equilibrium power operation." Proposed SR 3.4.6.1, which contains proposed requirements for sampling reactor coolant to determine specific activity, is modified by a note that requires this Surveillance to be performed only in MODE 1. This change is slightly more restrictive because sampling will be required whenever the reactor is in MODE 1 and not just when equilibrium conditions have been established. This change is consistent with NUREG-1433.
- M3 The Surveillance Frequency has been changed from monthly to weekly (every 7 days) for consistency with NUREG-1433, Rev. 1. Since Revision 1 to the NUREG deleted the surveillance requirement to verify that reactor coolant gross specific activity is less than or equal to 100/E-bar $\mu\text{Ci/gm}$ every 7 days, the reactor coolant specific activity trending interval was decreased to 7 days from 31 days.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

- LA1 CTS 4.6.B.6 contains requirements for reactor coolant and offgas system sampling during startup, following significant power level changes, and



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.6 - RCS SPECIFIC ACTIVITY

following significant changes in offgas radiation levels. The results of any of these samples are intended to determine if RCS specific activity is exceeding specified limits. Experience has determined that the weekly sampling required by proposed SR 3.4.6.1 and requirements for monitoring main steam line and offgas radiation levels is sufficient to ensure RCS specific activity levels are not exceeded. Therefore, RCS specific activity requirements for sampling stack gas, offgas and main steam line are being relocated to plant procedures and will be controlled in accordance with the licensee controlled programs. In addition, the criteria for when specific activity has been returned to limits (for 48 hours or until a stable iodine concentration below the limit has been established with at least 3 consecutive samples being taken in all cases) has been relocated to plant procedures and will be controlled by the licensee controlled programs. The method of determining dose equivalent I-131 (i.e., quantitative measurements of specific isotopes of Iodine), as described in CTS 4.6.B.5, has also been relocated to plant procedures. These changes are consistent with NUREG-1433.

"Specific"

- L1 Proposed ACTION A allows the LCO limit to be exceeded for 48 hours provided that the specific activity does not exceed 26 $\mu\text{Ci/gm}$. CTS 3.6.B.6 allows the limit to be exceeded during a power transient and limits the time the reactor can be operated, when the LCO RCS Specific Activity limit is exceeded, to less than 5% of its yearly power operation. Generic Letter 85-19, "Reporting Requirements on Primary Coolant Iodine Spikes," states that this limit is not necessary because reactor fuel has improved significantly since this requirement was established, and that proper fuel management by licensees and existing reporting requirements for fuel failures will preclude ever approaching this limit. Removal of this limit is consistent with the BWR/4 Standard Technical Specifications, NUREG-1433, requirements.
- L2 CTS 3.6.B.6 requires the reactor to be shut down and the steam line isolation valves to be closed immediately if the iodine concentration exceeds 26 $\mu\text{Ci/gm}$. Proposed ACTION B allows 12 hours to close the isolation valves or to be in Mode 3. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam isolation valves, or to achieve the required plant conditions, in an orderly manner and without challenging plant systems. The less restrictive 12 hour Completion Time is consistent with NUREG-1433.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

(A1)

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To assure the integrity and safe operation of the reactor coolant system.

Specification

~~A. Thermal and Pressurization Limitations~~

- LCD 3.4.9 1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification

(A1) ~~A. Thermal and Pressurization Limitations~~

SR 3.4.9.1

1. During heatups and cooldowns, the following parameters shall be recorded and reactor coolant temperature determined at 15-minute intervals until 3 successive readings at each given location are within 5°F.

(L1)

(30)

(LA1)

- a. Steam Dome Pressure (Convert to upper vessel region temperature)
- b. Reactor bottom drain temperature
- c. Recirculation loops A and B
- d. Reactor vessel bottom head temperature
- e. Reactor vessel shell adjacent to shell flange

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.A. Thermal and Pressurization Limitations~~

LCO
3.4.9

2. During all operations with a critical core, other than for low-level physics tests, except when the vessel is vented, the reactor vessel shell and fluid temperatures shall be at or above the temperature of curve #3 of Figure 3.6-1.

3. During heatup by nonnuclear means, except when the vessel is vented or as indicated in 3.6.A.4, during cooldown following nuclear shutdown, or during low-level physics tests, the reactor vessel temperature shall be at or above the temperatures of curve #2 of Figure 3.6-1 until removing tension on the head stud bolts as specified in 3.6.A.5.

LCO
3.4.9

~~4.6.A. Thermal and Pressurization Limitations~~

SR 3.4.9.1

2. Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange shall be recorded at least every ~~15-30~~ (L2) minutes during inservice hydrostatic or leak testing when the vessel pressure is > 312 psig.

Note 1 to
SR 3.4.9.1

(A2)

3. Test specimens representing the reactor vessel, base weld and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-82.

SEP 13 1995

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.A. Thermal and Pressurization Limitations~~

~~4.6.A. Thermal and Pressurization Limitations~~

AL

4. The beltline region of reactor vessel temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve #1 of Figure 3.6-1. The applicability of this curve to these tests is extended to nonnuclear heatup and ambient loss cooldown associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.

LCO 3.4.9

SR 3.4.9.1, Note 2

5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 80°F, and must remain above 80°F while under full tension.

SR 3.4.9.5 Note 2

LCO 3.4.9

~~4. DELETED~~

SR 3.4.9.1, Note 1

M1

Proposed SR 3.4.9.2

SR 3.4.9.5 + Note 1
SR 3.4.9.6 + Note
SR 3.4.9.7 + Note

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

A6

LA1

Proposed frequencies for SRs 3.4.9.5, 6+7

M2



MAR 24 1993

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~3.6.A Thermal and Pressurization Limitations (Cont'd)~~

6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other. (A4)

LCD 3.4.9

7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

LCD 3.4.9

(M3)

Proposed Actions
A, B + C

~~SURVEILLANCE REQUIREMENTS~~

~~A.6.A Thermal and Pressurization Limitations (Cont'd)~~

SR 3.4.9.4 + Note 1

6. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

(A3)

(LAI)

(A4)

SR 3.4.9.3 + Note

7. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

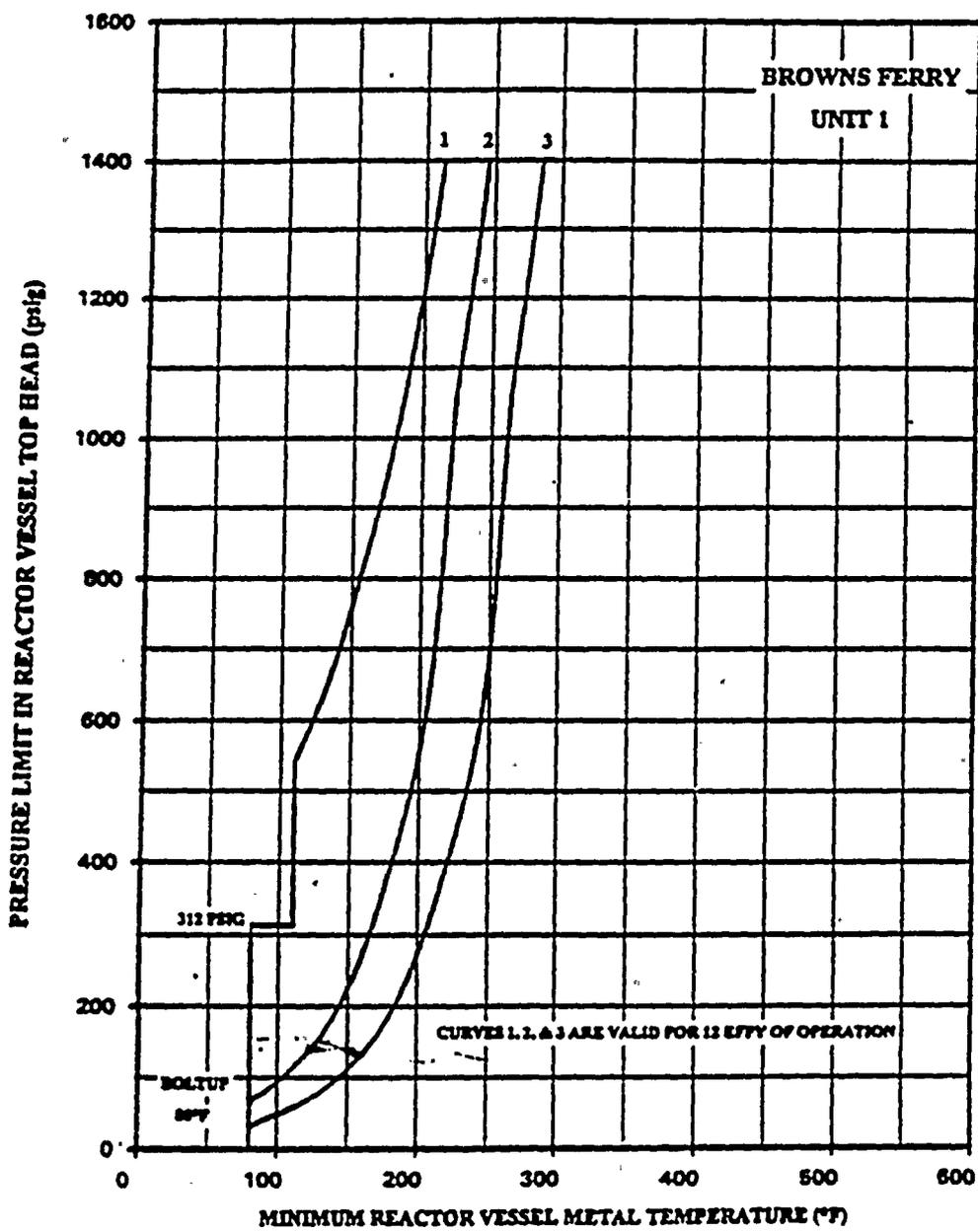
(LAI)

SEP 13 1995

3.4.9-1

Figure 3.6-1

(A1)



Curve No. 1
 Minimum temperature for pressure tests such as required by Section XI. Minimum temperature of 191°F is required for test pressure of 1,100 psig.

Curve No. 2
 Minimum temperature for mechanical heatup or cooldown following nuclear shutdown.

Curve No. 3
 Minimum temperature for core operation (criticality) includes additional margin required for 10CFR50, Appendix G, Par. IV.A.3 which became effective July 26, 1983.

Notes
 These curves include sufficient margin to provide protection against feedwater nozzle degradation. The curves allow for shifts in RT_{NDT} of the Reactor vessel belline materials, in accordance with Reg. Guide 1.89 Rev. 2, to compensate for radiation embrittlement for 12 EFPY.

3.4.9-1 (A1)
 Figure 3.6-1

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

See Justification for Changes for BFN 1STS 3.4.2

See Justification for Changes for BFN 1STS 3.4.1

4.6.E. Jet Pumps

- 2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

3.6.F Recirculation Pump Operation

- 1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.
- 2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.
- 3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation

SR 3.4.9.4 Note 2

4.6.F Recirculation Pump Operation

- 1. Recirculation pump speeds shall be checked and logged at least once per day.
- 2. No additional surveillance required.

SR 3.4.9.4

(LAI)

- 3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and ~~log~~ the loop discharge temperature and dome saturation temperature.



MAY 31 1994

~~3.6/A.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.F Recirculation Pump Operation

~~3.6.F.3 (Cont'd)~~

3.4.9.4, Note 2

temperature of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

See Justification for Changes for BFN 1STS 3.4.1

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

See Justification for Changes for CTS 3.6.G/4.6.G

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.



AL

3.6.B. Coolant Chemistry

4.6.B. Coolant Chemistry

1. PRIOR TO STARTUP and at steaming rates less than 100,000 lb/hr, the following limits shall apply.

- a. Conductivity, $\mu\text{mho/cm}$ at 25°C 2.0
- b. Chloride, ppm 0.1

1. Reactor coolant shall be continuously monitored for conductivity except when there is no fuel in the reactor vessel.

- a. Whenever the continuous conductivity monitor is inoperable, a sample of reactor coolant shall be analyzed for conductivity every 4 hours except as listed below. If the reactor is in COLD SHUTDOWN CONDITION, a sample of reactor coolant shall be analyzed for conductivity every 8 hours.
- b. Once a week the continuous monitor shall be checked with an in-line flow cell. This in-line conductivity calibration shall be performed every 24 hours whenever the reactor coolant conductivity is $>1.0 \mu\text{mho/cm}$ at 25°C.

2. At steaming rates greater than 100,000 lb/hr, the following limits shall apply.

- a. Conductivity, $\mu\text{mho/cm}$ at 25°C 1.0
- b. Chloride, ppm 0.2

2. During startup prior to pressurizing the reactor above atmospheric pressure, measurements of reactor water quality shall be performed to show conformance with 3.6.B.1 of limiting conditions.

(R1) →

(A1)

3.6.B. Coolant Chemistry

3. At steaming rates greater than 100,000 lb/hr, the reactor water quality may exceed Specification 3.6.B.2 only for the time limits specified below. Exceeding these time limits or the following maximum quality limits shall be cause for placing the reactor in the **COLD SHUTDOWN CONDITION**.

a. Conductivity time above 1 μ ho/cm at 25°C - 2 weeks/year.
Maximum Limit 10 μ ho/cm at 25°C

b. Chloride concentration time above 0.2 ppm - 2 weeks/year.
Maximum Limit - 0.5 ppm.

c. The reactor shall be placed in the **SHUTDOWN CONDITION** if pH <5.6 or >8.6 for a 24-hour period.

(R1)

4.6.B. Coolant Chemistry

3. Whenever the reactor is operating (including **HOT STANDBY CONDITION**) measurements of reactor water quality shall be performed according to the following schedule:

a. Chloride ion content and pH shall be measured at least once every 96 hours.

b. Chloride ion content shall be measured at least every 8 hours whenever reactor conductivity is >1.0 μ ho/cm at 25°C.

c. A sample of reactor coolant shall be measured for pH at least once every 8 hours whenever the reactor coolant conductivity is >1.0 μ ho/cm at 25°C.



JUN 28 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.B. Coolant Chemistry

4. When the reactor is not pressurized with fuel in the reactor vessel, except during the STARTUP CONDITION, the reactor water shall be maintained within the following limits.

- a. Conductivity - 10 μ ho/cm at 25°C
- b. Chloride - 0.5 ppm
- c. pH shall be between 5.3 and 8.6.

5. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated immediately. The reactor shall be brought to the COLD SHUTDOWN CONDITION as rapidly as cooldown rate permits.

6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 μ Ci/gm of dose equivalent I-131.

See Justification for changes to BFN ISTS 3.4.6 in this section

4.6.B. Coolant Chemistry

4. Whenever the reactor is not pressurized with fuel in the reactor vessel, a sample of the reactor coolant shall be analyzed at least every 96 hours for conductivity, chloride ion content and pH.

5. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least I-131, I-132, I-133, and I-134 shall be performed monthly on a coolant liquid sample.

6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:

(R1)

(AI)

3.6.B. Coolant Chemistry

3.6.B.6 (Cont'd)

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 $\mu\text{Ci/gm}$ whenever the reactor is critical. The reactor shall not be operated more than 5% of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 $\mu\text{Ci/gm}$, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

See Justification for Changes for BFN ISTS 3.4.6 in this section

(R1)

7. When there is no fuel in the reactor vessel, technical specification reactor coolant chemistry limits do not apply.

(R1)

7. When there is no fuel in the reactor vessel, sampling of reactor coolant chemistry at technical specification frequency is not required.

** For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.

4.6.B. Coolant Chemistry

4.6.B.6 (Cont'd)

- a. During the STARTUP CONDITION
- b. Following a significant power change**
- c. Following an increase in the equilibrium off-gas level exceeding 10,000 $\mu\text{Ci/sec}$ (at the steam jet air ejector) within a 48-hour period.
- d. Whenever the equilibrium iodine limit specified in 3.6.B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 $\mu\text{Ci/gm}$) is established. However, at least 3 consecutive samples shall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration.



MAY 31 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

temperature of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

See Justification for Changes to BFN 1STS 3.4.1, Recirculation Loops operating, in this section

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

(R1)

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.



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~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.G Structural Integrity

3.6.G.1 (Cont'd)

(R1) →

b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.

4.6.G. Structural Integrity

(R1) →

- 3. For Unit 1 an augmented inservice surveillance program shall be performed to monitor potential corrosive effects of chloride residue released during the March 22, 1975 fire. The augmented inservice surveillance program is specified as follows:
 - a. Browns Ferry Mechanical Maintenance Instruction 53, dated September 22, 1975, paragraph 4, defines the liquid penetrant examinations required during the first, second, third and fourth refueling outages following the fire restoration.
 - b. Browns Ferry Mechanical Maintenance Instruction 46, dated July 18, 1975, Appendix B, defines the liquid penetrant examinations required during the sixth refueling outage following the fire restoration.

JUL 05 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(AI)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.H. Snubbers

During all modes of operation, all snubbers shall be OPERABLE except as noted in 3.6.H.1. All safety-related snubbers are listed in Plant Surveillance Instructions.

1. With one or more snubber(s) inoperable on a system that is required to be OPERABLE in the current plant condition, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system inoperable and follow the appropriate Limiting Condition statement for that system.

(LAI)

4.6.H. Snubbers

Each safety-related snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 3.6.H/4.6.H. These snubbers are listed in Plant Surveillance Instructions.

1. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

2. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.6.H-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.6.H-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment No. 210.



(A1)

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4.6.H. Snubbers

3. Visual Inspection
Acceptance Criteria

Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.6.H.5. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the LIMITING CONDITIONS FOR OPERATION shall be met.

(LAI)



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

JUL 05 1994

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

AD

4.6.H Snubbers

4.6.H.3 (Cont'd)

LAI

Additionally, snubbers attached to sections of safety-related systems that have experienced unexpected potentially damaging transients since the last inspection period shall be evaluated for the possibility of concealed damage and functionally tested, if applicable, to confirm OPERABILITY. Snubbers which have been made inoperable as the result of unexpected transients, isolated damage, or other random events, when the provisions of 4.6.H.7 and 4.6.H.8 have been met and any other appropriate corrective action implemented, shall not be counted in determining the next visual inspection interval.

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

JUL 05 1994

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

4.6.H. Snubbers

4. FUNCTIONAL TEST Schedule, Lot Size, and Composition

During each refueling outage, a representative sample of 10% of the total of each type of safety-related snubbers in use in the plant shall be functionally tested either in place or in a bench test.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of snubbers within the types. The representative sample should be weighed to include more snubbers from severe service areas such as near heavy equipment.

The stroke setting and the security of fasteners for attachment of the snubbers to the component and to the snubber anchorage shall be verified on snubbers selected for FUNCTIONAL TESTS.

(2A1)



(A1)

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(LA1)

4.6.H. Snubbers

5. FUNCTIONAL TEST Acceptance Criteria

The snubber FUNCTIONAL TEST shall verify that:

- a. Activation (restraining action) is achieved in both tension and compression within the specified range, except that inertia dependent, acceleration limiting mechanical snubbers may be tested to verify only that activation takes place in both directions of travel.
- b. Snubber bleed, or release where required, is present in both compression and tension within the specified range.
- c. For mechanical snubbers, the force required to initiate or maintain motion of the snubber is not great enough to overstress the attached piping or component during thermal movement, or to indicate impending failure of the snubber.
- d. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.



JUL 05 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

4.6.H. Snubbers

4.6.H.5 (Cont'd)

e. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

6. FUNCTIONAL TEST Failure Analysis and Additional Test/Lots

(LAI)

An engineering evaluation shall be made of each failure to meet the FUNCTIONAL TEST acceptance criteria to determine the cause of the failure. The result of this analysis shall be used, if applicable, in selecting snubbers to be tested in the subsequent lot in an effort to determine the OPERABILITY of other snubbers which may be subject to the same failure mode. Selection of snubbers for future testing may also be based on the failure analysis. For each snubber that does not meet the FUNCTIONAL TEST acceptance criteria, an additional lot equal to 10 percent of the remainder of that type of snubbers shall be functionally tested. Testing shall continue until no additional inoperable snubbers are found within subsequent lots or all snubbers of the original FUNCTIONAL TEST type have been tested or all suspect snubbers identified by the failure analysis have been tested, as applicable.



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4.6.H. Snubbers

4.6.H.6 (Cont'd)

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency, all snubbers of the same design subject to the same defect shall be functionally tested.

This testing requirement shall be independent of the requirements stated above for snubbers not meeting the FUNCTIONAL TEST acceptance criteria.

The discovery of loose or missing attachment fasteners will be evaluated to determine whether the cause may be localized or generic. The result of the evaluation will be used to select other suspect snubbers for verifying the attachment fasteners, as applicable.

7. FUNCTIONAL TEST Failure - Attached Component Analysis

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are restrained by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components restrained by the

LAI →

AMENDMENT NO. 163



~~2.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

JAN 19 1989

~~LIMITING CONDITIONS FOR OPERATION~~~~SURVEILLANCE REQUIREMENTS~~

4.6.H. Snubbers

4.6.H.7 (Cont'd)

snubber(s) were adversely affected by the inoperability of the snubber(s), and in order to ensure that the restrained component remains capable of meeting the designed service.

8. Functional Testing Of Repaired and Spare Snubbers

Snubbers which fail the visual inspection or the FUNCTIONAL TEST acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the FUNCTIONAL TEST results shall meet the FUNCTIONAL TEST criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the FUNCTIONAL TEST must have been performed within 12 months before being installed in the unit.

9. Exemption from Visual Inspection or FUNCTIONAL TESTS

Permanent or other exemptions from visual inspections and/or functional testing for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and if applicable snubber life destructive testing was performed to qualify snubber OPERABILITY for the applicable design conditions at either the

(2A1)



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JAN 19 1989

4.6.H. Snubbers

4.6.H.9 (Cont'd)

completion of their fabrication or at a subsequent date. Snubbers so exempted shall continue to be listed in the plant instructions with footnotes indicating the extent of the exemptions.

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10. Snubber Service Life Program

The service life of snubbers may be extended based on an evaluation of the records of FUNCTIONAL TESTS, maintenance history, and environmental conditions to which the snubbers have been exposed.



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Table 4.6.H-1

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SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS		
	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or more	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.



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Table 4.6.H-1 (Continued)

SNUBBER VISUAL INSPECTION INTERVAL

- Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.
- Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C, but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.
- Note 6: The provisions of Specification 1.0.II are applicable for all inspection intervals up to and including 48 months.



Section 3.4, Reactor Coolant System (RCS) Bases

The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of the proposed Browns Ferry Unit 2 Technical Specification Section 3.4, consistent with the BWR Standard Technical Specification, NUREG 1433. The revised Bases are as shown in the proposed Browns Ferry Unit 2 Technical Specification Bases.

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



AI

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To assure the integrity and safe operation of the reactor coolant system.

Specification

A. Thermal and Pressurization Limitations

- LC0 3.4.9 1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification

AI

A. Thermal and Pressurization Limitations

SR 3.4.9.1

- Note to SR 3.4.9.1
1. During heatups and cooldowns, the following parameters shall be recorded and reactor coolant temperature determined at 15-minute intervals until 3 successive readings at each given location are within 5°F.

L1 30

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- a. Steam Dome Pressure (Convert to upper vessel region temperature)
- b. Reactor bottom drain temperature
- c. Recirculation loops A and B
- d. Reactor vessel bottom head temperature
- e. Reactor vessel shell adjacent to shell flange



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.A. Thermal and Pressurization Limitations~~

(A1)

~~4.6.A. Thermal and Pressurization Limitations~~

SR 3.4.9.1

LCO
3.4.9

2. During all operations with a critical core, other than for low-level physics tests, except when the vessel is vented, the reactor vessel shell and fluid temperatures shall be at or above the temperature of curve #3 of Figure 3.6-1.

2. Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange shall be recorded at least every 15 30' - L2 minutes during inservice hydrostatic or leak testing when the vessel pressure is > 312 psig.

Note 1 to SR 3.4.9.1

LCO
3.4.9

3. During heatup by nonnuclear means, except when the vessel is vented or as indicated in 3.6.A.4, during cooldown following nuclear shutdown, or during low-level physics tests, the reactor vessel temperature shall be at or above the temperatures of curve #2 of Figure 3.6-1 until removing tension on the head stud bolts as specified in 3.6.A.5.

(A2)

3. Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-82.



SEP 13 1995

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.A. Thermal and Pressurization Limitations (Cont'd)~~

~~4.6.A. Thermal and Pressurization Limitations (Cont'd)~~

4. The beltline region of reactor vessel temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve #1 of Figure 3.6-1. The applicability of this curve to these tests is extended to nonnuclear heatup and ambient loss cooldown associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.

LCO 3.4.9

SR 3.4.9.1, Note 2

SR 3.4.9.5 Note 2

LCO 3.4.9

(A1)

4. ~~>DELETED~~

SR 3.4.9.1, Note 1.

(M1)

Proposed SR 3.4.9.2

SR 3.4.9.5 + Note 1
SR 3.4.9.6 + Note
SR 3.4.9.7 + Note

5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 82°F, and must remain above 82°F while under full tension.

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

(LAI)

(AG)

Proposed frequencies for SRs 3.4.9.5, 6 + 7

(M2)



MAR 24 1993

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~3.6.A Thermal and Pressurization Limitations (Cont'd)~~

LCO 3.4.9 6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other. -

LCO 3.4.9 7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

M3 Proposed ACTIONS A, B + C

~~SURVEILLANCE REQUIREMENTS~~

~~4.6.A Thermal and Pressurization Limitations (Cont'd)~~

SR 34.9.4 + Note 1

A1 A3 6. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged. LA1

SR 3.4.9.3

Note

A4 7. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged. LA1



LIMITING CONDITIONS FOR OPERATION

(AI)

SURVEILLANCE REQUIREMENTS

See Justification for Changes for BFN ISTS 3.4.2

See Justification for Changes for BFN ISTS 3.4.1

4.6.E. Jet Pumps

2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

3.6.F. Recirculation Pump Operation

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.
2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.
3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval, restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature of the reactor

SR 3.4.9.4 Note 2

4.6.F. Recirculation Pump Operation

1. Recirculation pump speeds shall be checked and logged at least once per day.
2. No additional surveillance required.

SR 3.4.9.4

(LAI)

3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.

MAR 18 1993

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

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~~SURVEILLANCE REQUIREMENTS~~

3.6.F Recirculation Pump Operation

~~3.6.F.3 (Cont'd)~~

3.4.9.4, Note 2
vessel water as determined
by dome pressure. The
total elapsed time in natural
circulation and one pump
operation must be no greater
than 24 hours.

- 4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

See Justification for Change for BAN ISTS 3.4.1

3.6.G Structural Integrity

- 1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a COLD SHUTDOWN CONDITION or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

4.6.G Structural Integrity

- 1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
- 2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

See Justification for Changes for CTS 3.6.G/4.6.G

3.6/4.6-13

BNF
Unit 2

AMENDMENT NO. 206

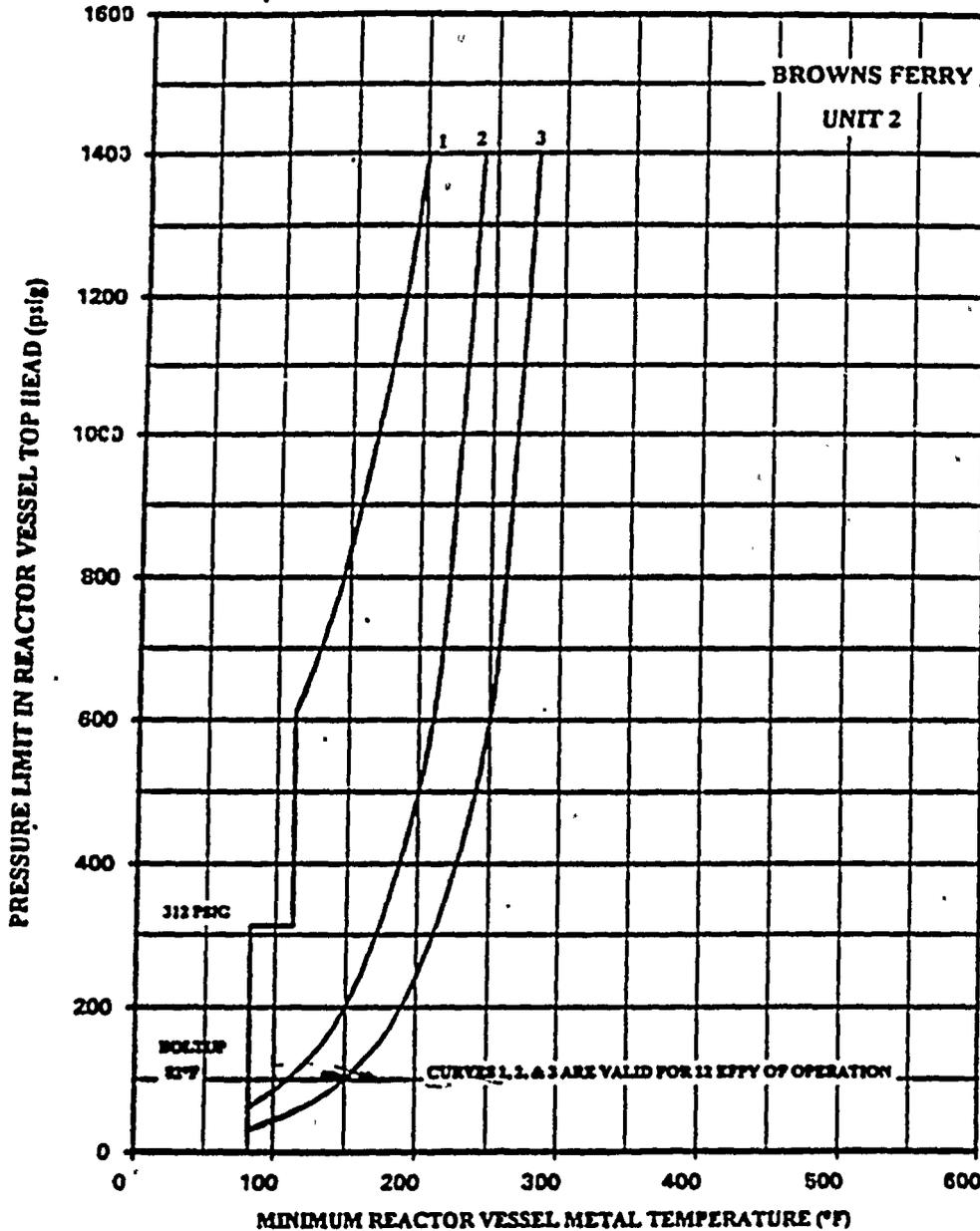


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3.4.9-1

Figure 3.6-1



Curve No. 1

Minimum temperature for pressure tests such as required by Section XI. Minimum temperature of 178°F is required for test pressure of 1,100 psig.

Curve No. 2

Minimum temperature for mechanical heatup or cooldown following nuclear shutdown.

Curve No. 3

Minimum temperature for core operation (criticality) includes additional margin required for 10CFR50, Appendix G, Par. IV.A.3 which became effective July 26, 1983.

Notes

These curves include sufficient margin to provide protection against feedwater nozzle degradation. The curves allow for shifts in RT_{NDT} of the Reactor vessel belline materials, in accordance with Reg. Guide 1.99 Rev. 2, to compensate for radiation embrittlement for 12 EFFY.

3.4.9-1

(A1)

Figure 3.6-1



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3.6.B. Coolant Chemistry

1. PRIOR TO STARTUP and at steaming rates less than 100,000 lb/hr, the following limits shall apply.

a. Conductivity, $\mu\text{mho/cm}$ at 25°C 2.0

b. Chloride, ppm 0.1

2. At steaming rates greater than 100,000 lb/hr, the following limits shall apply.

a. Conductivity, $\mu\text{mho/cm}$ at 25°C 1.0

b. Chloride, ppm 0.2

RI

4.6.B. Coolant Chemistry

1. Reactor coolant shall be continuously monitored for conductivity except when there is no fuel in the reactor vessel.

a. Whenever the continuous conductivity monitor is inoperable, a sample of reactor coolant shall be analyzed for conductivity every 4 hours except as listed below. If the reactor is in COLD SHUTDOWN CONDITION, a sample of reactor coolant shall be analyzed for conductivity every 8 hours.

b. Once a week the continuous monitor shall be checked with an in-line flow cell. This in-line conductivity calibration shall be performed every 24 hours whenever the reactor coolant conductivity is $>1.0 \mu\text{mho/cm}$ at 25°C.

2. During startup prior to pressurizing the reactor above atmospheric pressure, measurements of reactor water quality shall be performed to show conformance with 3.6.B.1 of limiting conditions.

AD

3.6.B. Coolant Chemistry

3. At steaming rates greater than 100,000 lb/hr, the reactor water quality may exceed Specification 3.6.B.2 only for the time limits specified below. Exceeding these time limits or the following maximum quality limits shall be cause for placing the reactor in the **GOLD SHUTDOWN CONDITION**.
 - a. Conductivity time above 1 $\mu\text{mho/cm}$ at 25°C - 2 weeks/year.
Maximum Limit 10 $\mu\text{mho/cm}$ at 25°C
 - b. Chloride concentration time above 0.2 ppm - 2 weeks/year.
Maximum Limit - 0.5 ppm.
 - c. The reactor shall be placed in the **SHUTDOWN CONDITION** if pH <5.6 or >8.6 for a 24-hour period.

4.6.B. Coolant Chemistry

3. Whenever the reactor is operating (including **HOT STANDBY CONDITION**) measurements of reactor water quality shall be performed according to the following schedule:
 - a. Chloride ion content and pH shall be measured at least once every 96 hours.
 - b. Chloride ion content shall be measured at least every 8 hours whenever reactor conductivity is >1.0 $\mu\text{mho/cm}$ at 25°C.
 - c. A sample of reactor coolant shall be measured for pH at least once every 8 hours whenever the reactor coolant conductivity is >1.0 $\mu\text{mho/cm}$ at 25°C.

(R1)



JUN 28 1994

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3.6.B. Coolant Chemistry

4.6.B. Coolant Chemistry

R1

- 4. When the reactor is not pressurized with fuel in the reactor vessel, except during the STARTUP CONDITION, the reactor water shall be maintained within the following limits.
 - a. Conductivity - 10 μ mho/cm at 25°C
 - b. Chloride - 0.5 ppm
 - c. pH shall be between 5.3 and 8.6.
- 5. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated immediately. The reactor shall be brought to the COLD SHUTDOWN CONDITION as rapidly as cooldown rate permits.
- 6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 μ Ci/gm of dose equivalent I-131.

- 4. Whenever the reactor is not pressurized with fuel in the reactor vessel, a sample of the reactor coolant shall be analyzed at least every 96 hours for conductivity, chloride ion content and pH.
- 5. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least I-131, I-132, I-133, and I-134 shall be performed monthly on a coolant liquid sample.
- 6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:

SEE JUSTIFICATION FOR CHANGES TO BFN ISTS 3.4.6 IN THIS SECTION



JUN 28 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

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~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.B. Coolant Chemistry

3.6.B.6 (Cont'd)

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 $\mu\text{Ci/gm}$ whenever the reactor is critical. The reactor shall not be operated more than 5% of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 $\mu\text{Ci/gm}$, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

SEE JUSTIFICATION FOR CHANGES FOR BFN 1STS 3.4.6 IN THIS SECTION

RI

7. When there is no fuel in the reactor vessel, technical specification reactor coolant chemistry limits do not apply.

4.6.B. Coolant Chemistry

4.6.B.6 (Cont'd)

- a. During the STARTUP CONDITION
- b. Following a significant power change**
- c. Following an increase in the equilibrium off-gas level exceeding 10,000 $\mu\text{Ci/sec}$ (at the steam jet air ejector) within a 48-hour period.
- d. Whenever the equilibrium iodine limit specified in 3.6.B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 $\mu\text{Ci/gm}$) is established. However, at least 3 consecutive samples shall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration.

RI

7. When there is no fuel in the reactor vessel, sampling of reactor coolant chemistry at technical specification frequency is not required.

** For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.

MAR 18 1993

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

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~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

See Justification for Changes to BFN 15TS 3.4.1, Recirculation Loops Operating, in this Section.

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a COLD SHUTDOWN CONDITION or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

RI

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.



MAR 18 1993

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

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~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.G Structural Integrity

3.6.G.1 (Cont'd)

(R1)

- b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.



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~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.H. Snubbers

During all modes of operation, all snubbers shall be OPERABLE except as noted in 3.6.H.1. All safety-related snubbers are listed in Plant Surveillance Instructions.

1. With one or more snubber(s) inoperable on a system that is required to be OPERABLE in the current plant condition, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system inoperable and follow the appropriate Limiting Condition statement for that system.

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4.6.H. Snubbers

Each safety-related snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 3.6.H/4.6.H. These snubbers are listed in Plant Surveillance Instructions.

1. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

2. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.6.H-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.6.H-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment No. 225.

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~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

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~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

4.6.H. Snubbers

3. Visual Inspection
Acceptance Criteria

Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY; (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.6.H.5. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the LIMITING CONDITIONS FOR OPERATION shall be met.

(LAI) →



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~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(AI)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

4.6.H Snubbers

4.6.H.3 (Cont'd)

(LAI)

~~Additionally, snubbers attached to sections of safety-related systems that have experienced unexpected potentially damaging transients since the last inspection period shall be evaluated for the possibility of concealed damage and functionally tested, if applicable, to confirm OPERABILITY. Snubbers which have been made inoperable as the result of unexpected transients, isolated damage, or other random events, when the provisions of 4.6.H.7 and 4.6.H.8 have been met and any other appropriate corrective action implemented, shall not be counted in determining the next visual inspection interval.~~



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4.6.H. Snubbers

4. FUNCTIONAL TEST Schedule, Lot Size, and Composition

During each refueling outage, a representative sample of 10% of the total of each type of safety-related snubbers in use in the plant shall be functionally tested either in place or in a bench test.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of snubbers within the types. The representative sample should be weighed to include more snubbers from severe service areas such as near heavy equipment.

The stroke setting and the security of fasteners for attachment of the snubbers to the component and to the snubber anchorage shall be verified on snubbers selected for FUNCTIONAL TESTS.

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4.6.H. Snubbers

5. FUNCTIONAL TEST Acceptance Criteria

The snubber FUNCTIONAL TEST shall verify that:

(LAI) →

- a. Activation (restraining action) is achieved in both tension and compression within the specified range, except that inertia dependent, acceleration limiting mechanical snubbers may be tested to verify only that activation takes place in both directions of travel.
- b. Snubber bleed, or release where required, is present in both compression and tension within the specified range.
- c. For mechanical snubbers, the force required to initiate or maintain motion of the snubber is not great enough to overstress the attached piping or component during thermal movement, or to indicate impending failure of the snubber.
- d. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.



JUL 05 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

AV

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

4.6.H. Snubbers

4.6.H.5 (Cont'd)

e. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

6. FUNCTIONAL TEST Failure Analysis and Additional Test Lots

LAI

An engineering evaluation shall be made of each failure to meet the FUNCTIONAL TEST acceptance criteria to determine the cause of the failure. The result of this analysis shall be used, if applicable, in selecting snubbers to be tested in the subsequent lot in an effort to determine the OPERABILITY of other snubbers which may be subject to the same failure mode. Selection of snubbers for future testing may also be based on the failure analysis. For each snubber that does not meet the FUNCTIONAL TEST acceptance criteria, an additional lot equal to 10 percent of the remainder of that type of snubbers shall be functionally tested. Testing shall continue until no additional inoperable snubbers are found within subsequent lots or all snubbers of the original FUNCTIONAL TEST type have been tested or all suspect snubbers identified by the failure analysis have been tested, as applicable.

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4.6.H. Snubbers

4.6.H.6 (Cont'd)

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If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency, all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the FUNCTIONAL TEST acceptance criteria.

The discovery of loose or missing attachment fasteners will be evaluated to determine whether the cause may be localized or generic. The result of the evaluation will be used to select other suspect snubbers for verifying the attachment fasteners, as applicable.

7. FUNCTIONAL TEST Failure - Attached Component Analysis

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are restrained by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components restrained by the

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4.6.H. Snubbers

4.6.H.7 (Cont'd)

snubber(s) were adversely affected by the inoperability of the snubber(s), and in order to ensure that the restrained component remains capable of meeting the designed service.

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8. Functional Testing Of Repaired and Spare Snubbers

Snubbers which fail the visual inspection or the FUNCTIONAL TEST acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the FUNCTIONAL TEST results shall meet the FUNCTIONAL TEST criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the FUNCTIONAL TEST must have been performed within 12 months before being installed in the unit.

9. Exemption from Visual Inspection or FUNCTIONAL TESTS

Permanent or other exemptions from visual inspections and/or functional testing for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and if applicable snubber life destructive testing was performed to qualify snubber OPERABILITY for the applicable design conditions at either the



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4.6.H. Snubbers

4.6.H.9 (Cont'd)

completion of their fabrication or at a subsequent date. Snubbers so exempted shall continue to be listed in the plant instructions with footnotes indicating the extent of the exemptions.

(LAI) →

10. Snubber Service Life Program

The service life of snubbers may be extended based on an evaluation of the records of FUNCTIONAL TESTS, maintenance history, and environmental conditions to which the snubbers have been exposed.



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Table 4.6.H-1
SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS		
	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or more	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.



JUL 05 1994

LAI

Table 4.6.H-1 (Continued)

SNUBBER VISUAL INSPECTION INTERVAL

- Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.
- Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C, but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.
- Note 6: The provisions of Specification 1.0.LL are applicable for all inspection intervals up to and including 48 months.

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To assure the integrity and safe operation of the reactor coolant system.

Specification

~~A. Thermal and Pressurization Limitations~~

LCO 3.4.9

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification

~~A. Thermal and Pressurization Limitations~~

(A1)

SR 3.4.9.1

Note: 1. to SR 3.4.9.1

(L1)

(30)

During heatups and cooldowns, the following parameters shall be recorded and reactor coolant temperature determined at 15-minute intervals until 3 successive readings at each given location are within 5°F.

(LA1)

- a. Steam Dome Pressure (Convert to upper vessel region temperature)
- b. Reactor bottom drain temperature
- c. Recirculation loops A and B
- d. Reactor vessel bottom head temperature
- e. Reactor vessel shell adjacent to shell flange

(A1)

~~3.6.A. Thermal and Pressurization Limitations~~

LCO
3.4.9

2. During all operations with a critical core, other than for low-level physics tests, except when the vessel is vented, the reactor vessel shell and fluid temperatures shall be at or above the temperature of curve #3 of Figure 3.6-1.

LCO
3.4.9

3. During heatup by nonnuclear means, except when the vessel is vented or as indicated in 3.6.A.4, during cooldown following nuclear shutdown, or during low-level physics tests, the reactor vessel temperature shall be at or above the temperatures of curve #2 of Figure 3.6-1 until removing tension on the head stud bolts as specified in 3.6.A.5.

~~4.6.A. Thermal and Pressurization Limitations~~

SR 3.4.9.1

2. Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange shall be recorded at least every ~~15-30~~ (L2) minutes during inservice hydrostatic or leak testing when the vessel pressure is > 312 psig.

Note 1 to SR 3.4.9.1

(A2)

3. Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-82.



(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.A. Thermal and Pressurization Limitations~~

~~A.6.A. Thermal and Pressurization Limitations~~

4. The beltline region of reactor vessel temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve #1 of Figure 3.6-1. The applicability of this curve to these tests is extended to nonnuclear heatup and ambient loss cooldown associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.

LCO 3.4.9
 SR 3.4.9.1
 Note 2

5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 70°F, and must remain above 70°F while under full tension.

SR 3.4.9.1
 Note 2
 LCO 3.4.9

~~A. DELETED~~

SR 3.4.9.1, Note 1

(M1) Proposed SR 3.4.9.2

SR 3.4.9.5 + Note 1
 SR 3.4.9.6 + Note
 SR 3.4.9.7 + Note

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

(A6)

(LA1)

Proposed Frequencies for SRs 3.4.9.5, 6 + 7

(M2)



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.6.A Thermal and Pressurization Limitations (Cont'd)~~ (A1)

LCO
3.4.9

6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

LCO
3.4.9

7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and bottom head drain are within 145°F.

(M3) Proposed ACTIONS
A, B + C

~~4.6.A Thermal and Pressurization Limitations (Cont'd)~~

SR3.4.9.4 + Note 1

6. Prior to and during STARTUP of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged. (A4) (LA1)

SR3.4.9.37.
+
Note

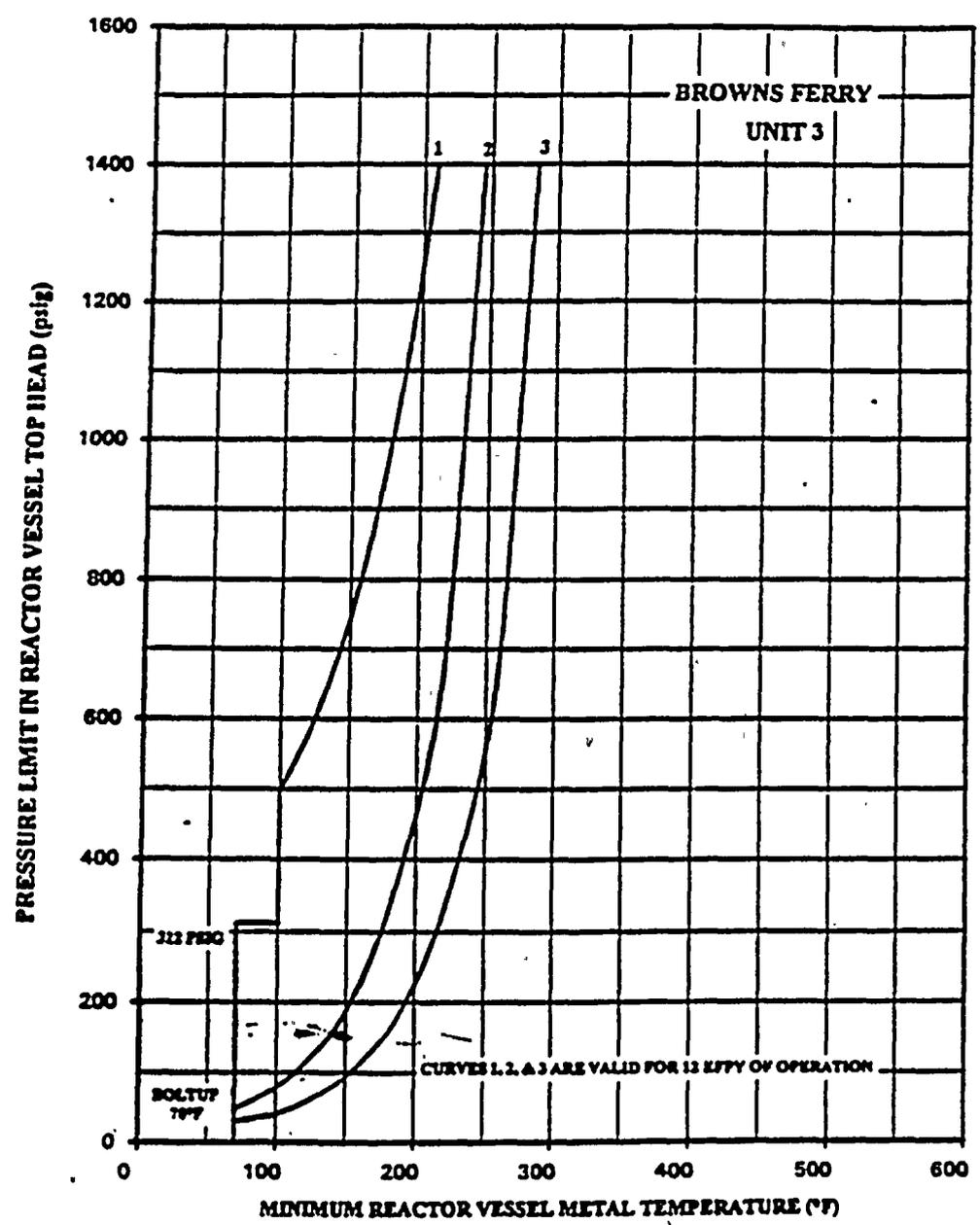
Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

(LA1)



SEP 13 1995

(A1) Figure 3.4.9-1
~~Figure 3.6-1~~



Curve No. 1
Minimum temperature for pressure tests such as required by Section XI. Minimum temperature of 189°F is required for test pressure of 1,100 psig.

Curve No. 2
Minimum temperature for mechanical heatup or cooldown following nuclear shutdown.

Curve No. 3
Minimum temperature for core operation (criticality) includes additional margin required for 10CFR50, Appendix G, Par. IV.A.3 which became effective July 26, 1983.

Notes
These curves include sufficient margin to provide protection against feedwater nozzle degradation. The curves allow for shifts in RT_{NDT} of the Reactor vessel beltline materials, in accordance with Reg. Guide 1.99 Rev. 2, to compensate for radiation embrittlement for 12 EFFY.

Figure 3.4.9(A1)
~~Figure 3.6-1~~



~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~
~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(AI)

See Justification for changes for BFN ISTS 3.4.2

See Justification for changes for BFN ISTS 3.4.1

4.6.E. Jet Pumps

- Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

3.6.F Recirculation Pump Operation

- The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.
- Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.
- When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature

4.6.F Recirculation Pump Operation

- Recirculation pump speeds shall be checked and logged at least once per day.
- No additional surveillance required.

SR 3.4.9.4

(LAL)

3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.

SR 4.9.4 of 2

(A1)

3.6.F Recirculation Pump Operation

~~3.6.F.3 (Cont'd)~~

3.4.9.4, Note 2

of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

- 4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

See Justification for Changes for BFN 1STS 3.4.1

3.6.G Structural Integrity

- 1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

4.6.G Structural Integrity

- 1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
- 2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

See Justification for Changes for CTS 3.6.G/4.6.G

(A1)

JUN 28 1994

3.6.B. Coolant Chemistry

4.6.B. Coolant Chemistry

- 1. PRIOR TO STARTUP and at steaming rates less than 100,000 lb/hr, the following limits shall apply.
 - a. Conductivity, $\mu\text{mho/cm}$ at 25°C 2.0
 - b. Chloride, ppm 0.1

- 2. At steaming rates greater than 100,000 lb/hr, the following limits shall apply.
 - a. Conductivity, $\mu\text{mho/cm}$ at 25°C 1.0
 - b. Chloride, ppm 0.2

- 1. Reactor coolant shall be continuously monitored for conductivity except when there is no fuel in the reactor vessel.
 - a. Whenever the continuous conductivity monitor is inoperable, a sample of reactor coolant shall be analyzed for conductivity every 4 hours except as listed below. If the reactor is in COLD SHUTDOWN CONDITION, a sample of reactor coolant shall be analyzed for conductivity every 8 hours.
 - b. Once a week the continuous monitor shall be checked with an in-line flow cell. This in-line conductivity calibration shall be performed every 24 hours whenever the reactor coolant conductivity is $>1.0 \mu\text{mho/cm}$ at 25°C.

- 2. During startup prior to pressurizing the reactor above atmospheric pressure, measurements of reactor water quality shall be performed to show conformance with 3.6.B.1 of limiting conditions.

(R1)



DEC 07 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

AD

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.B. Coolant Chemistry

3. At steaming rates greater than 100,000 lb/hr, the reactor water quality may exceed Specification 3.6.B.2 only for the time limits specified below. Exceeding these time limits or the following maximum quality limits shall be cause for placing the reactor in the COLD SHUTDOWN CONDITION.
- a. Conductivity time above 1 $\mu\text{mho/cm}$ at 25°C - 2 weeks/year.
Maximum Limit 10 $\mu\text{mho/cm}$ at 25°C
 - b. Chloride concentration time above 0.2 ppm - 2 weeks/year.
Maximum Limit - 0.5 ppm.
 - c. The reactor shall be placed in the SHUTDOWN CONDITION if pH <5.6 or >8.6 for a 24-hour period.

4.6.B. Coolant Chemistry

3. Whenever the reactor is operating (including HOT STANDBY CONDITION) measurements of reactor water quality shall be performed according to the following schedule:
- a. Chloride ion content and pH shall be measured at least once every 96 hours.
 - b. Chloride ion content shall be measured at least every 8 hours whenever reactor conductivity is >1.0 $\mu\text{mho/cm}$ at 25°C.
 - c. A sample of reactor coolant shall be measured for pH at least once every 8 hours whenever the reactor coolant conductivity is >1.0 $\mu\text{mho/cm}$ at 25°C.

RI



AD

3.6.B. Coolant Chemistry

4. When the reactor is not pressurized with fuel in the reactor vessel, except during the STARTUP CONDITION, the reactor water shall be maintained within the following limits.

R1

- a. Conductivity - 10 $\mu\text{mho/cm}$ at 25°C
- b. Chloride - 0.5 ppm
- c. pH shall be between 5.3 and 8.6.

5. When the time limits or maximum conductivity or chloride concentration limits are exceeded, an orderly shutdown shall be initiated immediately. The reactor shall be brought to the COLD SHUTDOWN CONDITION as rapidly as cooldown rate permits.

6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 $\mu\text{Ci/gm}$ of dose equivalent I-131.

See Justification for Changes to BFN ISTS 3.4.6 In this Section

4.6.B. Coolant Chemistry

4. Whenever the reactor is not pressurized with fuel in the reactor vessel, a sample of the reactor coolant shall be analyzed at least every 96 hours for conductivity, chloride ion content and pH.

5. During equilibrium power operation an isotopic analysis, including quantitative measurements for at least I-131, I-132, I-133, and I-134 shall be performed monthly on a coolant liquid sample.

6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:



AI

3.6.B. Coolant Chemistry

3.6.B.6 (Cont'd)

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 $\mu\text{Ci/gm}$ whenever the reactor is critical. The reactor shall not be operated more than 5% of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 $\mu\text{Ci/gm}$, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

See Justification for Changes BFN 15 TS 3.4.6 in this Section

RI

7. When there is no fuel in the reactor vessel, technical specification reactor coolant chemistry limits do not apply.

4.6.B. Coolant Chemistry

4.6.B.6 (Cont'd)

- a. During the STARTUP CONDITION
- b. Following a significant power change**
- c. Following an increase in the equilibrium off-gas level exceeding 10,000 $\mu\text{Ci/sec}$ (at the steam jet air ejector) within a 48-hour period.
- d. Whenever the equilibrium iodine limit specified in 3.6.B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 $\mu\text{Ci/gm}$) is established. However, at least 3 consecutive samples shall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration.

RI

7. When there is no fuel in the reactor vessel, sampling of reactor coolant chemistry at technical specification frequency is not required.

** For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.



MAY 31 1994

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(AI)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

see justification for Changes to BFN ISTS 3.4.1, Recirculation Loops operating, in this section

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.

(RI)

- a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components 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 NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.6.G Structural Integrity

3.6.G.1 (Cont'd)

(R1)

- b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.

4.6.G Structural Integrity



JUL 05 1994

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3.6.H. Snubbers

During all modes of operation, all snubbers shall be OPERABLE except as noted in 3.6.H.1. All safety-related snubbers are listed in Plant Surveillance Instructions.

1. With one or more snubber(s) inoperable on a system that is required to be OPERABLE in the current plant condition, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system inoperable and follow the appropriate Limiting Condition statement for that system.

1
|

4.6.H. Snubbers

Each safety-related snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 3.6.H/4.6.H. These snubbers are listed in Plant Surveillance Instructions.

1. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

2. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.6.H-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.6.H-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment No. 183.

LAI



AI

4.6.H. Snubbers

3. Visual Inspection Acceptance Criteria

Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.6.H.5. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the LIMITING CONDITIONS FOR OPERATION shall be met.

LAI

(A1)

4.6.H Snubbers

4.6.H.3 (Cont'd)

Additionally, snubbers attached to sections of safety-related systems that have experienced unexpected potentially damaging transients since the last inspection period shall be evaluated for the possibility of concealed damage and functionally tested, if applicable, to confirm OPERABILITY. Snubbers which have been made inoperable as the result of unexpected transients, isolated damage, or other random events, when the provisions of 4.6.H.7 and 4.6.H.8 have been met and any other appropriate corrective action implemented, shall not be counted in determining the next visual inspection interval.

(LAI)





4.6.H. Snubbers

4. FUNCTIONAL TEST Schedule, Lot Size, and Composition

During each refueling outage, a representative sample of 10% of the total of each type of safety-related snubbers in use in the plant shall be functionally tested either in place or in a bench test.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of snubbers within the types. The representative sample should be weighed to include more snubbers from severe service areas such as near heavy equipment.

The stroke setting and the security of fasteners for attachment of the snubbers to the component and to the snubber anchorage shall be verified on snubbers selected for FUNCTIONAL TESTS.

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~~3.6/4.6 PRIMARY SYSTEM BOUNDARY~~

(AI)

~~LIMITING CONDITIONS FOR OPERATION~~~~SURVEILLANCE REQUIREMENTS~~4.6.H. Snubbers5. FUNCTIONAL TEST Acceptance Criteria

The snubber FUNCTIONAL TEST shall verify that:

- a. Activation (restraining action) is achieved in both tension and compression within the specified range, except that inertia dependent, acceleration limiting mechanical snubbers may be tested to verify only that activation takes place in both directions of travel.
- b. Snubber bleed, or release where required, is present in both compression and tension within the specified range.
- c. For mechanical snubbers, the force required to initiate or maintain motion of the snubber is not great enough to overstress the attached piping or component during thermal movement, or to indicate impending failure of the snubber.
- d. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

(LAI) →



(A1)

4.6.H. Snubbers

4.6.H.5 (Cont'd)

c. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

6. FUNCTIONAL TEST Failure Analysis and Additional Test Lots

(LAI) →

An engineering evaluation shall be made of each failure to meet the FUNCTIONAL TEST acceptance criteria to determine the cause of the failure. The result of this analysis shall be used, if applicable, in selecting snubbers to be tested in the subsequent lot in an effort to determine the OPERABILITY of other snubbers which may be subject to the same failure mode. Selection of snubbers for future testing may also be based on the failure analysis. For each snubber that does not meet the FUNCTIONAL TEST acceptance criteria, an additional lot equal to 10 percent of the remainder of that type of snubbers shall be functionally tested. Testing shall continue until no additional inoperable snubbers are found within subsequent lots or all snubbers of the original FUNCTIONAL TEST type have been tested or all suspect snubbers identified by the failure analysis have been tested, as applicable.



AI

4.6.H. Snubbers

4.6.H.6 (Cont'd)

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency, all snubbers of the same design subject to the same defect shall be functionally tested.

This testing requirement shall be independent of the requirements stated above for snubbers not meeting the FUNCTIONAL TEST acceptance criteria.

The discovery of loose or missing attachment fasteners will be evaluated to determine whether the cause may be localized or generic. The result of the evaluation will be used to select other suspect snubbers for verifying the attachment fasteners, as applicable.

7. FUNCTIONAL TEST Failure - Attached Component Analysis

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are restrained by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components restrained by the

LAI

(A1)

4.6.H. Snubbers

4.6.H.7 (Cont'd)

snubber(s) were adversely affected by the inoperability of the snubber(s), and in order to ensure that the restrained component remains capable of meeting the designed service.

8. Functional Testing Of Repaired and Spare Snubbers

(LAI)

Snubbers which fail the visual inspection or the FUNCTIONAL TEST acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the FUNCTIONAL TEST results shall meet the FUNCTIONAL TEST criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the FUNCTIONAL TEST must have been performed within 12 months before being installed in the unit.

9. Exemption from Visual Inspection or FUNCTIONAL TESTS

Permanent or other exemptions from visual inspections and/or functional testing for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and if applicable snubber life destructive testing was performed to qualify snubber OPERABILITY for the applicable design conditions at either the

(A1)

4.6.H. Snubbers

4.6.H.9 (Cont'd)

completion of their fabrication or at a subsequent date. Snubbers so exempted shall continue to be listed in the plant instructions with footnotes indicating the extent of the exemptions.

(LAI) →

10. Snubber Service Life Program

The service life of snubbers may be extended based on an evaluation of the records of FUNCTIONAL TESTS, maintenance history, and environmental conditions to which the snubbers have been exposed.



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BFN
Unit 3

3.6/4.6-23a

AMENDMENT NO. 134

LA1

Table 4.6.H-1

JUL 05 1994

SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS		
	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
50	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or more	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Column A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

JUL 05 1994

Table 4.6.H-1 (Continued)

SNUBBER VISUAL INSPECTION INTERVAL

- Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.
- Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C, but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.
- Note 6: The provisions of Specification 1.0.LL are applicable for all inspection intervals up to and including 48 months.

LAI

PAGE 18 OF 18



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

ADMINISTRATIVE

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 These surveillances are a duplication of the regulations found in 10 CFR 50 Appendix H. These regulations require licensee compliance and can not be revised by the licensee. Therefore, these details of the regulations within the Technical Specifications are repetitious and unnecessary. Furthermore, approved exemptions to the regulations, and exceptions presented within the regulations themselves, are also details which are adequately presented without repeating the details within the Technical Specifications. Therefore, retaining the requirement to meet the requirements of 10 CFR 50 Appendix H, as modified by approved exemptions, and eliminating the Technical Specification details that are also found in Appendix H, is considered a presentation preference which is administrative in nature.

- A3 For clarity, the terms "prior to and during startup" and "prior to" have been replaced with "15 minutes". This Frequency is effectively the same since the proposed Surveillance now must be performed no more than 15 minutes prior to startup of the idle recirculation loop. This is essentially equivalent to the current requirements.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

- A4 Proposed SR 3.4.9.4 requires verification that the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature are within 50°F of each other. CTS 3.6.A.6/4.6.A.6 requires verification that the temperatures between the idle and operating recirculation loops are within 50°F of each other. The temperature of the "operating recirculation loop" is considered equivalent to the RPV temperature. Therefore, this change is considered administrative.
- A6 Proposed SRs 3.4.9.5, 6 & 7 require the reactor vessel flange and head flange temperatures be verified > 82°F, while CTS 4.6.A.5 requires the reactor vessel shell temperature immediately below the head flange be recorded. The BFN procedure that implements this requirement requires the vessel flange and head flange temperature be verified and requires the shell temperature be recorded. Since the intent of the surveillance is to verify vessel flange and head flange temperature to satisfy CTS 3.6.A.5 and both the current and the proposed SRs do this, the two are considered equivalent. As such, the proposed change is administrative.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 A new Surveillance Requirement has been added. SR 3.4.9.2 ensures the RCS pressure and temperature are within the criticality limits once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality. This is an additional restriction on plant operation.
- M2 Three new Surveillance Requirements have been added. SR 3.4.9.5 ensures the vessel head is not tensioned at too low a temperature every 30 minutes. SRs 3.4.9.6 and 3.4.9.7 ensure the vessel and head flange temperatures do not exceed the minimum allowed temperature. These are additional restrictions on plant operation since the current requirements have no times specified.
- M3 ACTIONS have been added (proposed ACTIONS A, B, and C) to provide direction when the LCO is not met. Currently, no real ACTIONS are provided. These ACTIONS are consistent with the BWR Standard Technical Specification, NUREG 1433, and are additional restrictions on plant operation.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LAI Details of the methods for performing Surveillances, and any requirement to record data, are relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 The Frequency of this Surveillance has been changed from 15 minutes to 30 minutes. Verification that RCS temperature is within limits every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes is reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits a reasonable time for assessment and correction of minor deviations. In addition, this new Frequency is consistent with the BWR Standard Technical Specification, NUREG 1433.
- L2 The Frequency of this Surveillance has been changed from 15 minutes to 30 minutes. The metal temperature is not expected to change rapidly due to its large mass, thus a 30 minute Frequency is adequate. In addition, this new Frequency is consistent with the BWR Standard Technical Specification, NUREG 1433.

PAGE 3 OF 7



JUSTIFICATION FOR CHANGES
CTS 3.6.B/4.6.B - COOLANT CHEMISTRY

RELOCATED SPECIFICATIONS

R1 The chemistry limits are provided to prevent long term component degradation and provide long term maintenance of acceptable structural conditions of the system. The associated surveillances are not required to ensure immediate operability of the reactor coolant system. Therefore, the requirements specified in current Specification 3.6.B/4.6.B did not satisfy the NRC Final Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the Browns Ferry Unit 2 Technical Specifications and have been relocated to plant documents controlled in accordance with 10CFR50.59.

JUSTIFICATION FOR CHANGES
CTS 3.6.G/4.6.G - STRUCTURAL INTEGRITY

RELOCATED SPECIFICATIONS

RI The structural integrity inspections are provided to prevent long term component degradation and provide long term maintenance of acceptable structural conditions of the system. The associated inspections are not required to ensure immediate operability of the system. Therefore, the requirements specified in current Specification 3.6.G/4.6.G did not satisfy the NRC Final Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the BFN Unit 2 Technical Specifications and have been relocated to plant documents controlled in accordance with 10CFR50.59.



JUSTIFICATION FOR CHANGES
CTS 3.6.H/4.6.H - SNUBBERS

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LAI Snubber inspection requirements are part of the BFN Inservice Inspection (ISI) Program and are being relocated to the ISI program documents. Requirements for the ISI Program are specified in 10 CFR 50.55a to be performed in accordance with ASME Section XI. NRC regulations contain the necessary programmatic requirements for ISI without repeating them in the proposed BFN ISTS. Changes to the ISI Program are controlled in accordance with 10 CFR 50.59. With the removal of operability requirements from the Technical Specifications, snubber operability requirements will be determined in accordance with Technical Specification system operability requirements.

Section 3.4, Reactor Coolant System (RCS) Bases

The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of the proposed Browns Ferry Unit 2 Technical Specification Section 3.4, consistent with the BWR Standard Technical Specification, NUREG 1433. The revised Bases are as shown in the proposed Browns Ferry Unit 2 Technical Specification Bases.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



NOV 22 1988

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

(A1)

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and containment cooling systems.

Objective

To assure the OPERABILITY of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

~~A. Core Spray System (CSS)~~

(A1)

LCO 3.5.1

1. The CSS shall be OPERABLE:

(1) PRIOR TO STARTUP from a COLD CONDITION, or

(A6)

(2) when there is irradiated fuel in the vessel and when the reactor vessel pressure is greater than atmospheric pressure, except as specified in Specification 3.5.A.2.

Applicability

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the surveillance requirements of the core and containment cooling systems when the corresponding limiting condition for operation is in effect.

Objective

To verify the OPERABILITY of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

~~A. Core Spray System (CSS)~~

(A2)

1. Core Spray System Testing.

(L1) Item Actual or (A3) Frequency
SR 3.5.1.9 a. Simulated Automatic Actuation test
Once/18 mos Operating Cycle

Proposed Note to SR 3.5.1.9

(A4) SR 3.5.1.6 b. Pump OPERABILITY Per Specification 1.0.MM

(A9) c. Motor Operated Valve OPERABILITY Per Specification 1.0.MM

SR 3.5.1.6 d. System flow rate: Each loop shall deliver at least 6250 gpm against a system head corresponding to a (A3) Once/3-months (92 days)



AUG 02 1989

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.A Core Spray System (CSS)~~

(A1)

~~4.5.A Core Spray System (CSS)~~

~~4.5.A.1.d (Cont'd)~~

2. ACTION A
 ACTION H
 If one CSS loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are OPERABLE.

(A5)

(M4) Be in Mode 3 in 12hrs

3. ACTION B & H
 If Specification 3.5.A.1 or Specification 3.5.A.2 cannot be met, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(36) (L2)

4. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one OPERABLE pump and associated diesel generator shall be OPERABLE, except with the reactor vessel head removed as specified in 3.5.A.5 or PRIOR TO STARTUP as specified in 3.5.A.1.

(A6)

See Justification for Changes for BFN 15TS 3.5.2

SR 3.5.1.6
 105 psi differential pressure between the reactor vessel and the primary containment.

(A9) e. Check Valve Per Specification 1.0.MM

~~SR 3.5.1.2~~

f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct position.

(31 days)

Once/Month

(A3)

(A1)

(A7)

2. No additional surveillance is required.

See Justification for Changes for BFN 15TS 3.8.1

(A7)

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~ (A1)

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~ (L1) (A3)

(A2) 1. The RHRS shall be OPERABLE #:

- (1) PRIOR TO STARTUP from a COLD CONDITION; or (A6)
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

Applicability

1. a. Actual or Simulated Automatic Actuation Test (A3) Once/18 months Operating Cycle

SR 3.5.1.6 b. Pump OPERABILITY (A3) Per Specification 1.0.MM

(A9) c. Motor Operated valve OPERABILITY (A3) Per Specification 1.0.MM

SR 3.5.1.6 d. Pump Flow Rate (A3) Once/3 months (92 days)

(A9) e. Test Check Valve (A3) Per Specification 1.0.MM (31 days)

SR 3.5.1.2 f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct position. (A7) (A3) (31 days)

SR 3.5.1.4 g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. (A3) (A7) (31 days)

SR 3.5.1.2 Note

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

(A7) * Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

AUG 02 1989

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENT~~

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B.1 (cont'd)~~
SR 3.5.1.6

M3

2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.

2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

See Justification for Changes for BFN 1STS 3.6.2.4

ACTION A

ACTION H

3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.

(A1)

~~3. No additional surveillance required.~~

(A5)

See Justification for Changes for BFN 1STS 3.8.1

ACTION B

4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(A1)

~~4. No additional surveillance required.~~

36

L2

M4

in Mode 3 in 12 hrs and



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

ACTIONS
B+H

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.
36 L2

(A1) 8. No additional surveillance required.

Be in Mode 3 in 12 hrs

(M4)

See Justification for Changes for BFN ISTS 3.5.2

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

(R1)

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

AUG 02 1989

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

SURVEILLANCE REQUIREMENTS

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(RI)

- 12. If one RHR pump or associated heat exchanger located on the unit cross-connection in the adjacent unit is inoperable for any reason (including valve inoperability, pipe break, etc.), the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pump and associated diesel generator are OPERABLE.
- 13. If RHR cross-connection flow or heat removal capability is lost, the unit may remain in operation for a period not to exceed 10 days unless such capability is restored.

(LAS)

- 14. All recirculation pump discharge valves shall be OPERABLE PRIOR TO STARTUP (or closed if permitted elsewhere in these specifications).

- 12. No additional surveillance required.
- 13. No additional surveillance required.

SR 3.5.1.5

Note for SR 3.5.1.5

- 14. All recirculation pump discharge valves shall be tested for OPERABILITY during any period of COLD SHUTDOWN CONDITION exceeding 48 hours, if OPERABILITY tests have not been performed during the preceding 31 days.

~~3.9/4.9 AUXILIARY ELECTRICAL SYSTEM~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.9.A. Auxiliary Electrical Equipment~~

~~4.9.A. Auxiliary Electrical System~~

See Justification for Changes for BFN ISTS Section 3.3.8.1

See Justification for Changes for BFN ISTS 3.8.7

5. Logic Systems

a. Common accident signal logic system is OPERABLE.

See Justification for Changes for BFN ISTS 3.8.1

b. 480-V load shedding logic system is OPERABLE.

6. There shall be a minimum of 35,280 gallons of diesel fuel in each of the 7-day diesel-generator fuel tank assemblies.

See Justification for Changes for BFN ISTS 3.8.3

4.9.A.4. (Cont'd)

c. The loss of voltage and degraded voltage relays which start the diesel generators from the 4-kV shutdown boards shall be calibrated annually for trip and reset and the measurements logged. These relays shall be calibrated as specified in Table 4.9.A.4.c.

d. 4-kV shutdown board voltages shall be recorded once every 12 hours.

5. 480-V RMOV Boards 1D and 1E

SR 3.5.1.12

a. Once per ^{12 mo.} operating cycle the automatic transfer feature for 480-V RMOV boards 1D and 1E shall be functionally tested to verify auto-transfer capability. (A3)

~~3.9/4.9 AUXILIARY ELECTRICAL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.9.A. Auxiliary Electrical Equipment~~

~~4.9.A. Auxiliary Electrical System~~

~~3.9.A.3 (Cont'd)~~

- d. The 480-V shutdown boards 1A and 1B are energized.
- e. The units 1 and 2 diesel auxiliary boards are energized.

See Justification for Changes for BFN ISTS 3.8.7

- f. Loss of voltage and degraded voltage relays OPERABLE on 4-kV shutdown boards A, B, C, and D.

See Justification for Changes for BFN ISTS 3.3.8.1

- g. Shutdown buses 1 and 2 energized.

See Justification for Changes for BFN ISTS 3.8.1

(A13)

- h. The 480-V reactor motor-operated valve (RMOV) boards 1D & 1E are energized with motor-generator (mg) sets 1DN, 1DA, 1EN, and 1EA in service.

- 4. The three 250-V unit batteries, the four shutdown board batteries, a battery charger for each battery, and associated battery boards are OPERABLE.

See Justification for Changes for BFN ISTS 3.8.4 and 3.8.7

4. Undervoltage Relays

- a. (Deleted)
- b. Once every 18 months, the conditions under which the loss of voltage and degraded voltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator will start.

~~3.9/4.9 AUXILIARY ELECTRICAL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

A1

~~SURVEILLANCE REQUIREMENTS~~

~~3.9.B. Operation With Inoperable Equipment~~

12. When one 480-V shutdown board is found to be INOPERABLE, the reactor will be placed in HOT STANDBY CONDITION within 12 hours and COLD SHUTDOWN CONDITION within 24 hours.

See Justification for Changes for BFN 15TS 3.8.7

13. If one 480-V RMOV board mg set is INOPERABLE, REACTOR POWER OPERATION may continue for a period not to exceed seven days, provided the remaining 480-V RMOV board mg sets and their associated loads remain OPERABLE.

A13

14. If any two 480-V RMOV board mg sets become INOPERABLE, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

15. If the requirements for operating in the conditions specified by 3.9.B.1 through 3.9.B.14 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

See Justification for Changes for BFN 15TS Section 3.8

FEB 07 1991

~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

3.5.D Equipment Area Coolers

1. The equipment area cooler associated with each RHK pump and the equipment area cooler associated with each set of core spray pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.
2. When an equipment area cooler is not OPERABLE, the pump(s) served by that cooler must be considered inoperable for Technical Specification purposes.

4.5.D Equipment Area Coolers

1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

(LAS)

~~F. High Pressure Coolant Injection System (HPCIS)~~ (A2)

LCO 3.5.1

Applicability (A11)

Proposed Note for SR 3.5.1.8

1. The HPCI system shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in Specification 3.5.E.2. OPERABILITY shall be determined within 12 hours after reactor steam pressure reaches 150 psig from a GOLD CONDITION, or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

(LAG)

(A1)

~~F. High Pressure Coolant Injection System (HPCIS)~~

1. HPCI Subsystem testing shall be performed as follows:

SR 3.5.1.9 a. ^(Actual or) Simulated Automatic Actuation Test (L1) Once/18 months

(A4) SR 3.5.1.7 b. Pump OPERABILITY (A3) Per Specification 1.0.MM

(L11) (A9) c. Motor Operated Valve OPERABILITY (A3) Per Specification 1.0.MM

SR 3.5.1.7 d. Flow Rate at normal reactor vessel operating pressure (A3) Once/3-months (92 days) 920 to 1010 psig (L9)

Proposed Note for SR 3.5.1.7

(A10)

FEB 07 1991

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.E High Pressure Coolant Injection System (HPCIS)~~

~~4.5.E High Pressure Coolant Injection System (HPCIS)~~

~~4.5.E.1 (Cont'd)~~

SR 3.5.1.8 e. Flow Rate at Once/18
150 psig months

≤ 165 psig
(L10)

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

SR 3.5.1.2

f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct position.

31 days
Once/Month
(A3)

(A1) ~~2. No additional surveillances are required.~~

2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ~~ABS, CSS, RHRS~~ (LPCI), and RCICS are OPERABLE.

(L4)

verified immediately

Action H (A5)
Proposed Action D (L3)

3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

(A12)

62 in 3 in 12 hrs

(L2)

ACTIONS G+H

(A7)

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

(M4)

F. Reactor Core Isolation Cooling System (RCICS)

F. Reactor Core Isolation Cooling System (RCICS)

1. The RCICS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

1. RCIC Subsystem testing shall be performed as follows:
a. Simulated Auto- Once/18
matic Actuation months
Test

BFN Unit 1

See Justification for Changes 3.5/4.5-14 for BFN ISTS 3.5.3

AMENDMENT NO. 180



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.G Automatic Depressurization System (ADS)~~

~~4.5.G Automatic Depressurization System (ADS)~~ (A3)

(A2) 1. Six valves of the Automatic Depressurization System shall be OPERABLE:

1. During each operating cycle the following tests shall be performed on the ADS:

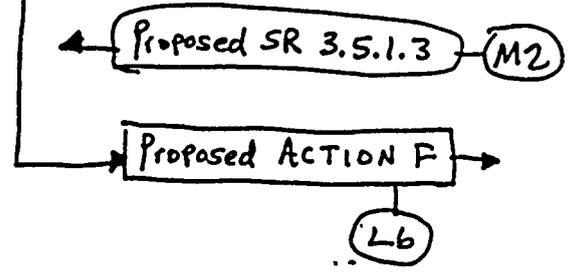
Applicability (A11) (1) PRIOR TO STARTUP from a COLD CONDITION, or, (L5) (150) (2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.G.2 and 3.5.G.3 below.

SR 3.5.1.10 a. (L1) Actual or (L1) A simulated automatic actuation test shall be performed PRIOR TO STARTUP after each refueling outage. (A3) Manual surveillance of the relief valves is covered in 4.6.D.2. (A8) (A4) Proposed Note to SR 3.5.1.10

ACTIONS F 2. With one of the above required ADS valves inoperable, provided the HPCI system, the core spray system, and the LPCI system are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least a HOT SHUTDOWN CONDITION within the next 12 hours and reduce reactor steam dome pressure to ≤ 105 psig within 24 hours. (36) (L2) (150) (L5)

(A1) 2. No additional surveillances are required.

ACTION G+H 3. With two or more of the above required ADS valves inoperable, be in at least a HOT SHUTDOWN CONDITION within 12 hours and reduce reactor steam dome pressure to ≤ 105 psig within 24 hours. (36) (L2) (150) (L5)



ACTIONS G Required Action H.1 (LEO 3.0.3) (M1) MODE 2 within 7hrs MODE 3 within 13hrs (L12) (for ADS only) MODE 4 within 37hrs (L12)

DEC 07 1994

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.C Coolant Leakage

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

3.6.D Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

See Justification for Changes for BFN 1STS 3.4.3

4.6.C Coolant Leakage

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

See Justification for Changes for BFN 1STS 3.4.5

4.6.D Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

SE 3.5.1.11

Proposed Note for FR 3.5.1.11

LB

(A1)

MAY 19 1994

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.H. Maintenance of Filled Discharge Pipe~~

~~4.5.H. Maintenance of Filled Discharge Pipe~~

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSS head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

LA2

See Justification for Changes for BFN 15 TS 3.5.3

SR 3.5.1.1

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

LA7

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined.

LA3

2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

LA4

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

LA3

4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

(A1)

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and containment cooling systems.

Objective

To assure the OPERABILITY of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

(A1)

LCO 3.5.1

1. The CSS shall be OPERABLE:

(A2)

(1) PRIOR TO STARTUP from a COLD CONDITION, or

(A6)

Applicability

(2) when there is irradiated fuel in the vessel and when the reactor vessel pressure is greater than atmospheric pressure, except as specified in Specification 3.5.A.2.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the surveillance requirements of the core and containment cooling systems when the corresponding limiting condition for operation is in effect.

Objective

To verify the OPERABILITY of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. Core Spray System Testing.

(L1)

Item

Frequency

SR 3.5.1.9

Actual or

(Simulated Automatic Actuation test

Once/18 mos. Operating Cycle

(A3)

Proposed Note to SR 3.5.1.9

(A4)

SR 3.5.1.6

b. Pump Operability

Per Specification 1.0.MM

(A3)

(A9)

c. Motor Operated Valve OPERABILITY

Per Specification 1.0.MM

SR 3.5.1.6

d. System flow rate: Each loop shall deliver at least 6250 gpm against a system head corresponding to a

Once/3-months

(A3)

92 days

2...

15



AUG 02 1989

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.A Core Spray System (CSS)

(A1)

2. If one CSS loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are OPERABLE.

ACTION A

ACTION H

M4 Be in Mode 3 in 12 hrs

3. If Specification 3.5.A.1 or Specification 3.5.A.2 cannot be met, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

ACTION B + H

4. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one OPERABLE pump and associated diesel generator shall be OPERABLE, except with the reactor vessel head removed as specified in 3.5.A.5 or PRIOR TO STARTUP as specified in 3.5.A.1.

See Justification for Changes for BFN ISTS 3.5.2

4.5.A Core Spray System (CSS)

4.5.A.1.d (Cont'd)

SR 3.5.1.6 105 psi differential pressure between the reactor vessel and the primary containment.

(A9) e. Check Valve Per Specification 1.0.MM

SR 3.5.1.2 1. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct position.

Once/Month (A3)

31 days

(A7)

(A1) 2. No additional surveillance is required.

See Justification for Changes for BFN ISTS 3.8.1

(A7) * Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

(A6)

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS APR 19 1994

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

A2 1. The RHRS shall be OPERABLE #:

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

Applicability

1. a. Actual or L1 (A3) Simulated Automatic Actuation Test Once/18 mos Operating Cycle

SR 3.5.1.6 b. Pump OPERABILITY Per Specification 1.0.MM (A3)

(A9) c. Motor Operated valve OPERABILITY Per Specification 1.0.MM

SR 3.5.1.6 d. Pump Flow Rate Once/3 months (A3) 92 days

(A9) e. Testable Check Valve Per Specification 1.0.MM

SR 3.5.1.2 f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct position. (A7) (A3) 31 days

SR 3.5.1.4 g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. (A3) (A7) 31 days

SR 3.5.1.2 Note

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

(A7) Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



AUG 02 1989

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENT

3.5.B ~~Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

4.5.B. ~~Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

4.5.B.1 (cont'd)

SR 3.5.1.6

M3

2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.

2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

See Justification for Changes for BFN 1STS 3.6.2.4

ACTION A

ACTION H

3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.

(A1) 3. No additional surveillance required.

(AS)

See Justification for Changes for BFN 1STS 2.8.1

ACTION B

4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(A1) 4. No additional surveillance required.

24
36

MIDE 3
M4 in 12 hrs



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

ACTIONS
B+H

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, ~~an orderly shutdown shall be initiated~~ and the reactor shall be placed in the COLD SHUTDOWN CONDITION within ~~24~~³⁶ hours. (L2)

~~4.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

~~8. No additional surveillance required.~~

Be in Mode 3 in 12 hrs (M4)

See Justification for Changes for BFN 1573 3.5.2

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

(R1) 11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

12. If three RHR pumps or associated heat exchangers located on the unit cross-connection in the adjacent units are inoperable for any reason (including valve inoperability, pipe break, etc.), the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pump and associated diesel generator are OPERABLE.

(R1)

13. If RHR cross-connection flow or heat removal capability is lost, the unit may remain in operation for a period not to exceed 10 days unless such capability is restored.

14. All recirculation pump discharge valves shall be OPERABLE PRIOR TO STARTUP (or closed if permitted elsewhere in these specifications).

(LAS)

~~12. No additional surveillance required.~~

~~13. No additional surveillance required.~~

SR 3.5.1.5

14. All recirculation pump discharge valves shall be tested for OPERABILITY - (during any period of COLD SHUTDOWN CONDITION exceeding 48 hours, if OPERABILITY tests have not been performed during the preceding 31 days.

Note for SR 3.5.1.5.

NOV 04 1991

~~3.9/4.9 AUXILIARY ELECTRICAL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.9.A Auxiliary Electrical Equipment~~

~~4.9.A Auxiliary Electrical System~~

~~3.9.A.3. (Cont'd)~~

- d. The 480-V shutdown boards 2A and 2B are energized.
- e. The units 1 and 2 diesel auxiliary boards are energized.

See Justification for Changes for BFN ISTS 3.8.7

- f. Loss of voltage and degraded voltage relays OPERABLE on 4-kV shutdown boards A, B, C, and D.

See Justification for Changes for BFN ISTS 3.3.8.1

- g. Shutdown buses 1 and 2 energized.

See Justification for Changes for BFN ISTS 3.8.1

- h. The 480-V reactor motor-operated valve (RMOV) boards 2D & 2E are energized with motor-generator (mg) sets 2DN, 2DA, 2EN, and 2EA in service.

(A13)

- 4. The three 250-V unit batteries, the four shutdown board batteries, a battery charger for each battery, and associated battery boards are OPERABLE.

See Justification for Changes for BFN ISTS 3.8.4

- 4. Undervoltage Relays
 - a. (Deleted)
 - b. Once every 18 months, the conditions under which the loss of voltage and degraded voltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator will start.

FEB 12 1991

~~3.9/4.9 AUXILIARY ELECTRICAL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.9.A. Auxiliary Electrical Equipment~~

~~4.9.A. Auxiliary Electrical System~~

See Justification for Changes for BFN ISTS Section 3.3.8.1

See Justification for Changes for BFN ISTS 3.8.7

5. Logic Systems

a. Common accident signal logic system is OPERABLE.

See Justification for Changes for BFN ISTS 3.8.1

b. 480-V load shedding logic system is OPERABLE.

6. There shall be a minimum of 35,280 gallons of diesel fuel in each of the 7-day diesel-generator fuel tank assemblies.

See Justification for Changes for BFN ISTS 3.8.3

4.9.A.4. (Cont'd)

c. The loss of voltage and degraded voltage relays which start the diesel generators from the 4-kV shutdown boards shall be calibrated annually for trip and reset and the measurements logged. These relays shall be calibrated as specified in Table 4.9.A.4.c.

d. 4-kV shutdown board voltages shall be recorded once every 12 hours.

5. 480-V RMOV Boards 2D and 2E

SR 3.5.1.12

a. Once per ^{18 mos.} operating cycle the automatic transfer feature for 480-V RMOV boards 2D and 2E shall be functionally tested to verify auto-transfer capability. (A3)

AMENDMENT NO. 191

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.9.B. Operation With Inoperable Equipment

A1

12. When one 480-V shutdown board is found to be INOPERABLE, the reactor will be placed in the HOT STANDBY CONDITION within 12 hours and COLD SHUTDOWN CONDITION within 24 hours.

See Justification for Changes for BFN ISTS 3.8.7

13. If one 480-V RMOV board mg set is INOPERABLE, REACTOR POWER OPERATION may continue for a period not to exceed seven days, provided the remaining 480-V RMOV board mg sets and their associated loads remain OPERABLE.

A13

14. If any two 480-V RMOV board mg sets become INOPERABLE, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

15. If the requirements for operating in the conditions specified by 3.9.B.1 through 3.9.B.14 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

See Justification for Changes for BFN ISTS Section 3.8



FEB 07 1991

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.5.D Equipment Area Coolers

1. The equipment area cooler associated with each RHR pump and the equipment area cooler associated with each set of core spray pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.
2. When an equipment area cooler is not OPERABLE, the pump(s) served by that cooler must be considered inoperable for Technical Specification purposes.

4.5.D Equipment Area Coolers

1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

LAS

~~E. High Pressure Coolant Injection System (HPCIS)~~

(A2)

LCO 3.5.1

Applicability

(All)

Proposed Note for SR 3.5.1.8

1. The HPCI system shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in Specification 3.5.E.2. OPERABILITY shall be determined within 12 hours after reactor steam pressure + flow reaches 150 psig from a COLD CONDITION, or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

LAG

(A1)

~~F. High Pressure Coolant Injection System (HPCIS)~~

SR 3.5.1.9

Proposed Note for SR 3.5.1.9

(A4)

SR 3.5.1.7

(L11)

Proposed Note for SR 3.5.1.7

(A10)

1. HPCI Subsystem testing shall be performed as follows:

Actual or (L1)

- a. Simulated Automatic Actuation Test

Once/18 months

- b. Pump OPERABILITY

Per Specification 1.0.MM (A3)

- c. Motor Operated Valve OPERABILITY

Per Specification 1.0.MM (A9)

- d. Flow Rate at normal reactor vessel operating pressure

Once/3-months (A3)

(92 days)

920 to 1010 psig (L9)



LIMITING CONDITIONS FOR OPERATION (A1) SURVEILLANCE REQUIREMENTS

3.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

SR 3.5.L8 e. Flow Rate at Once/18
150 psig months

≤ 165 psig
L10
The HPCI pump shall deliver at least 5000 gpm during each flow rate test. 31 days

SR 3.5.1.2 f. Verify that Once/Month
each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct position. A3

(A1) 2. No additional surveillances are required.

(A5) ACTION H

L4
2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 147 days, provided the CSS, RHRS (LPCI) and RCICS are OPERABLE. Proposed ACTION D

3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 36 hours. ACTIONS G+H

(A7)
* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

F. Reactor Core Isolation Cooling System (RCICS)

F. Reactor Core Isolation Cooling System (RCICS)

1. The RCICS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

1. RCIC Subsystem testing shall be performed as follows:
a. Simulated Auto- Once/18 matic Actuation months Test

BFN Unit 2

3.5/4.5-14

AMENDMENT NO. 190

See Justification for Changes for BFN 15TS 3.5.3



LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.G Automatic Depressurization System (ADS) A1

4.5.G Automatic Depressurization System (ADS) A3

1. Six valves of the Automatic Depressurization System shall be OPERABLE: A2

1. During each operating cycle the following tests shall be performed on the ADS: L1

(1) PRIOR TO STARTUP from a COLD CONDITION, or, A11

SR 3.5.1.10

a. A simulated automatic actuation test shall be performed PRIOR TO STARTUP after each refueling outage. A3

Applicability

(2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except in the COLD SHUT-DOWN CONDITION or as specified in 3.5.G.2 and 3.5.G.3 below. 150

every 18 mos.

Manual surveillance of the relief valves is covered in 4.6.D.2. A8

Proposed Note to SR 3.5.1.10 A4

2. With one of the above required ADS valves inoperable, provided the HPCI system, the core spray system, and the LPCI system are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least a HOT SHUTDOWN CONDITION within the next 12 hours and reduce reactor steam dome pressure to 105 psig within hours. 150 LS

2. No additional surveillances are required. A1

ACTIONS E

Proposed SR 3.5.1.3 M2

ACTION G+H

3. With two or more of the above required ADS valves inoperable, be in at least a HOT SHUTDOWN CONDITION within 12 hours and reduce reactor steam dome pressure to 105 psig within hours. 150 LS 24 36 L2

Proposed ACTION F L6

ACTIONS G

3. With two or more of the above required ADS valves inoperable, be in at least a HOT SHUTDOWN CONDITION within 12 hours and reduce reactor steam dome pressure to 105 psig within hours. 150 LS 24 36 L2

Required Action H.1 (CCO 3.03) M1

Mode 2 within 7 hrs

Mode 3 within 13 hrs

Mode 4 within 37 hrs L12

(for ADS only) L12

AMENDMENT NO. 190



3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.C Coolant Leakage

- Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

SR 89-77B/77A-2-039

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

- If the condition in 1. or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

3.6.D Relief Valves

- When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

See Justification for Changes for BFN 15TS 2.4.3

4.6.C Coolant Leakage

- With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

See Justification for Changes for BFN 15TS 3.4.5

4.6.D Relief Valves

- Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.
- In accordance with Specification 1.0.MM, each relief valve shall be manually opened until

thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

SR 3.5.1.11

Proposed Note for SR 3.5.1.11

(L8)

(LA3)



AUG 02 1989

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.5.H. Maintenance of Filled Discharge Pipe

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

LA2

See Justification for Changes for BFN 15TS 3.5.3

4.5.H. Maintenance of Filled Discharge Pipe

SR 3.5.1.1

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

LA7

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined.

LA3

2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

LA4

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

LAB

4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

UNIT 3
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



LIMITING CONDITIONS FOR OPERATION

A1

SURVEILLANCE REQUIREMENTS

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and containment cooling systems.

Objective

To assure the OPERABILITY of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

~~A. Core Spray System (CSS)~~

A1

LCO 3.5.1

1. The CSS shall be OPERABLE:

A2

(1) PRIOR TO STARTUP from a COLD CONDITION, or

A6

(2) when there is irradiated fuel in the vessel and when the reactor vessel pressure is greater than atmospheric pressure, except as specified in Specification 3.5.A.2.

Applicability

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the surveillance requirements of the core and containment cooling systems when the corresponding limiting condition for operation is in effect.

Objective

To verify the OPERABILITY of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

~~A. Core Spray System (CSS)~~

1. Core Spray System Testing.

Item Frequency

(L1) Actual or

a. Simulated Automatic Actuation test

Once/18 months Operating Cycle

SR 3.5.1.9

Proposed Note to SR 3.5.1.9

b. Pump OPERABILITY

Per Specification 1.0.MM

(A4) SR 3.5.1.6

c. Motor Operated Valve OPERABILITY

Per Specification 1.0.MM

(A9)

d. System flow rate: Each loop shall deliver at least 6250 gpm against a system head corresponding to a

Once/3 months

92 days

SR 3.5.1.6



LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.A Core Spray System (CSS)

(A1)

- 2. **Action A** If one CSS loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are OPERABLE.
- Action H**

M4

Be in Mode 3 IN 12hrs

Action B+H

- 3. If Specification 3.5.A.1 or Specification 3.5.A.2 cannot be met, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours. (36) (L2)

- 4. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one OPERABLE pump and associated diesel generator shall be OPERABLE, except with the reactor vessel head removed as specified in 3.5.A.5 or PRIOR TO STARTUP as specified in 3.5.A.1.

(A6)

See Justification for changes for BFN ISTS 3.5.2

4.5.A Core Spray System (CSS)

4.5.A.1.d (Cont'd)

SR 3.5.1.6 } 105 psi differential pressure between the reactor vessel and the primary containment.

(A9) e. Testable Per Check Valve Specification 1.0MM

SR 3.5.1.2

- f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct (A7) position. (A3) Once/Month 31 days

(A1) 2. No additional surveillance is required.

See Justification for Changes for BFN ISTS 3.8.1

(A7) * Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

(A1)

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A2) 1. The RHRS shall be OPERABLE #:

(1) PRIOR TO STARTUP from a COLD CONDITION; or (A6)

Applicability (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

(L13)

OR
verify the manual shutoff valve in the LPCI cross tie is closed

SR 3.5.1.2 Note

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

1. a. Actual OR (L1) (A3) Simulated Automatic Actuation Test Once/18^{mos} Operating Cycle
- SR 3.5.1.6 b. Pump OPERABILITY (A3) Per Specification 1.0.MM
- (A9) c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- SR 3.5.1.6 d. Pump Flow Rate (A3) Once/3 months (92 days)
- (A9) e. Testable Check Valve Per Specification 1.0.MM
- SR 3.5.1.2 f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct position. (A7) (A3) (31 days) Once/Month
- SR 3.5.1.4 8. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. (A7) (A3) (31 days) Once/Month

(A7) Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



AUG 02 1989

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

SURVEILLANCE REQUIREMENT

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B.1 (cont'd)~~

SR 3.5.1.6

M3

2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps—containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.

2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

See Justification for Changes for BFN ISTS 3.6.2.4

Action A

Action H

3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.

(A1)

~~3. No additional surveillance required.~~

(A5)

See Justification for Changes for BFN ISTS 3.8.1

Action B

4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(A1)

~~4. No additional surveillance required.~~

24
36

Be in mode 3 in 12 hrs

(L2)

(M4)



(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

Actions B+H

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, ~~an orderly shutdown shall be initiated~~ and the reactor shall be placed in the COLD SHUTDOWN CONDITION within ~~24~~ ³⁰ hours. (L2)

(A1)

~~8. No additional surveillance required.~~

Be in Mode 3 in 12 hrs

(M4)

See Justification For Changes for BFN ISTS 3.5.2

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

(R)

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The B and D RHR pumps on unit 2 which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0:MM when the cross-connect capability is required.



AUG 02 1989

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.B Residual Heat Removal System
(RHR) (LPCI and Containment
Cooling)~~

~~A.5.B Residual Heat Removal System
(RHR) (LPCI and Containment
Cooling)~~

(RI)

- 12. If one RHR pump or associated heat exchanger located on the unit cross-connection in unit 2 is inoperable for any reason (including valve inoperability, pipe break, etc.), the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pump and associated diesel generator are OPERABLE.
- 13. If RHR cross-connection flow or heat removal capability is lost, the unit may remain in operation for a period not to exceed 10 days unless such capability is restored.

- ~~12. No additional surveillance required.~~
- 13. No additional surveillance required.

(LAS)

- 14. All recirculation pump discharge valves shall be OPERABLE PRIOR TO STARTUP (or closed if permitted elsewhere in these specifications).

SR 3.5.15

Note for SR 3.5.15

- 14. All recirculation pump discharge valves shall be tested for OPERABILITY during any period of COLD SHUTDOWN CONDITION exceeding 48 hours, if OPERABILITY tests have not been performed during the preceding 31 days.



FEB 14 1995

~~3.9/4.9 AUXILIARY ELECTRICAL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

~~3.9.A. Auxiliary Electrical Equipment~~

~~4.9.A. Auxiliary Electrical System~~

See Justification for changes for BFN ISTS Section 3.3.8.1

See Justification for changes for BFN ISTS 3.8.7

- 5. Logic Systems
 - a. Accident signal logic system is OPERABLE.
 - b. 480-volt load shedding logic system is OPERABLE.
- See Justification for changes for BFN ISTS 3.8.1

6. There shall be a minimum of 35,280 gallons of diesel fuel in each of the 7-day diesel-generator fuel tank assemblies.

See Justification for changes for BFN ISTS 3.8.3

4.9.A.4. (Cont'd)

c. The loss of voltage and degraded voltage relays which start the diesel generators from the 4-kV shutdown boards shall be calibrated annually for trip and reset and the measurements logged. These relays shall be calibrated as specified in Table 4.9.A.4.c.

d. 4-kV shutdown board voltages shall be recorded once every 12 hours.

SR 3.5.1.12

- 5. 480-V RMOV Boards 3D and 3E (18 mos.) (A3)
- a. Once per operating cycle, the automatic transfer feature for 480-V RMOV boards 3D and 3E shall be functionally tested to verify auto-transfer capability.



(A1)

NOV 04 1991

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.9.A Auxiliary Electrical Equipment~~

~~4.9.A. Auxiliary Electrical System~~

~~3.9.A.3. (Cont'd)~~

e. Loss of voltage and degraded voltage relays OPERABLE on 4-kV shutdown boards 3EA, 3EB, 3EC, and 3ED.

See Justification for Changes for BFN ISTS 3.3.8.1

f. The 480-V diesel auxiliary boards 3EA and 3EB are energized.

See Justification for Changes for BFN ISTS 3.8.7

g. The 480-V reactor motor-operated valve (RMOV) boards 3D & 3E are energized with motor-generator (mg) sets 3DN, 3DA, 3EN, and 3EA in service.

(A13)

4. The 250-V shutdown board 3EB battery, all three unit batteries, a battery charger for each battery, and associated battery boards are OPERABLE.

4. Undervoltage Relays

- a. (Deleted)
- b. Once every 18 months, the conditions under which the loss of voltage and degraded voltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator will start.

See Justification for Changes for BFN ISTS 3.8.4 + 3.8.7

See Justification for Changes for BFN ISTS 3.8.1



~~3.9/4.9 AUXILIARY ELECTRICAL SYSTEM~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.9.B Operation With Inoperable Equipment~~

10. When one 480-V shutdown board is found to be inoperable, the reactor will be placed in HOT STANDBY CONDITION within 12 hours and COLD SHUTDOWN CONDITION within 24 hours.

Justification for changes for BFN 1STS 3.8.7

11. If one 480-V RMOV board mg set is inoperable, REACTOR POWER OPERATION may continue for a period not to exceed seven days, provided the remaining 480-V RMOV board mg sets and their associated loads remain OPERABLE.

(A13)

12. If any two 480-V RMOV board mg sets become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

13. If the requirements for operation in the conditions specified by 3.9.B.1 through 3.9.B.12 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

See Justification for Changes for BFN 1STS section 3.8



(A1)

3.5.D Equipment Area Coolers

1. The equipment area cooler associated with each RHR pump and the equipment area cooler associated with each set of core spray pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.
2. When an equipment area cooler is not OPERABLE, the pump(s) served by that cooler must be considered inoperable for Technical Specification purposes.

4.5.D Equipment Area Coolers

1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

(LA5)

~~F. High Pressure Coolant Injection System (HPCIS)~~

(A2)

LCO 3.5.1

Applicability

(A11)

1. The HPCI system shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in Specification 3.5.E.2. OPERABILITY shall be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION, or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

Proposed Note for SR 3.5.1.8

(LAG)

(A1)

~~F. High Pressure Coolant Injection System (HPCIS)~~

SR 3.5.1.9

Proposed Note for SR 3.5.1.9

(A4)

(L11)

1. HPCI Subsystem testing shall be performed as follows:

a. ^(Actual or) Simulated Automatic Actuation Test

Once/18 months

b. Pump OPERABILITY

Per Specification 1.0.MM

c. Motor Operated Valve OPERABILITY

Per Specification 1.0.MM

d. Flow Rate at normal reactor vessel operating pressure

Once/3-months

92 days

Proposed Note for SR 3.5.1.7

(A10)

920 to 1010 psig

(L9)



AI

3.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

SR 3.5.1.8

e. Flow Rate at Once/18 months

≤ 165 psig

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

L10

f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct position.

SR 3.5.1.2

2. No additional surveillances are required.

2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS (LPCI), and RCICS are OPERABLE immediately.

3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24-36 hours.

Proposed ACTION D

Actions G+H

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

F. Reactor Core Isolation Cooling System (RCICS)

F. Reactor Core Isolation Cooling System (RCICS)

1. The RCICS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

1. RCIC Subsystem testing shall be performed as follows: a. Simulated Auto- Once/18 matic Actuation months Test

BFN Unit 3

See Justification for Changes 3.5/4.5-14 for BFN ISTS 3.5.3

AMENDMENT NO. 152

MAY 19 1994

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.G Automatic Depressurization System (ADS)

4.5.G Automatic Depressurization System (ADS)

1. Six valves of the Automatic Depressurization System shall be OPERABLE:

(1) PRIOR TO STARTUP from a COLD CONDITION, or,
(2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.G.2 and 3.5.G.3 below.

Applicability
All

2. With one of the above required ADS valves inoperable, provided the HPCI system, the core spray system, and the LPCI system are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least a HOT SHUTDOWN CONDITION within the next 12 hours and reduce reactor steam dome pressure to ≤ 105 psig within 24 hours.

Actions
F

Action
G & H

3. With two or more of the above required ADS valves inoperable, be in at least a HOT SHUTDOWN CONDITION within 12 hours and reduce reactor steam dome pressure to ≤ 105 psig within 24 hours.

Action
G

(for ADS only)
L12

Required Action H1 (LCO 3.0.3)
mode 2 within 7 hrs
mode 3 within 13 hrs
mode 4 within 37 hours

1. During each operating cycle the following tests shall be performed on the ADS:

SR 3.5.1.10 a. A simulated automatic actuation test shall be performed PRIOR TO STARTUP after each refueling outage. Manual surveillance of the relief valves is covered in 4.6.D.2.

Proposed Note to SR 3.5.1.10

2. No additional surveillances are required.

Proposed SR 3.5.1.3

Proposed Action F

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.C Coolant Leakage

- 2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

- 3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

3.6.D Relief Valves

- 1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

See Justification for Changes for BFN ISTS 3.4.3

4.6.C Coolant Leakage

- 2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

See Justification for Changes for BFN ISTS 3.4.5

4.6.D Relief Valves

- 1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.
- 2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened

until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve

SR3.5.1.11

Proposed Note for SR 3.5.1.11

LB

AMENDMENT NO. 188

PAGE 14 OF 15



~~3.5.H Maintenance of Filled Discharge Pipe~~

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

LA2

See Justification for Changes for BFN 15 TS 3.5.3

~~4.5.H. Maintenance of Filled Discharge Pipe~~

SR 3.5.1.1

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

LA7

1. Every month and prior to the testing of the RHRs (LPCI and Containment Spray) and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow determined.

LA3

2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

LA4

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

LA3

4. When the RHRs and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 Five current LCOs, 3.5.A, 3.5.B, 3.5.E, 3.5.G, and 3.5.H, have been combined into one proposed LCO (3.5.1). As such, the new LCO combines the three ECCS spray/injection Systems (HPCI, LPCI, and CS) into one LCO statement. The Bases continue to describe what components make up an ECCS subsystem. The new LCO statement also specifies that the six ADS valves are required. In addition, the ADS valve cycling requirements located in current Specification 4.6.D.1 are included as part of ADS operability. Thus, if an ADS valve does not cycle, the affected ECCS system is considered inoperable and the appropriate ACTION taken.

- A3 The Frequencies of "Once/operating cycle," "during each operating cycle," and "after each refueling outage" have been changed to "18 months." This is considered equivalent since 18 months is the length of an operating cycle or a refueling outage cycle. The Frequencies of "Once/3 months" and "Per Specification 1.0.MM" have been changed to "92 days," or "In accordance with the Inservice Testing program" as appropriate. The IST program test frequency for pumps is every 3 months and is currently defined by Specification 1.0.MM. Therefore, this change is considered administrative in nature. The Frequency of "Once/month" has been changed to "31 days."

- A4 Notes allowing actual vessel injection or ADS valve actuation to be excluded from this test (simulated automatic actuation test) have been added to proposed SR 3.5.1.9 and SR 3.5.1.10. Since the current



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING

requirements state the test is "simulated" (i.e., valve actuation and vessel injection are inherently excluded), this allowance is considered administrative in nature.

- A5 Proposed Condition H provides direction for various interrelationships between HPCI and ADS, and between LPCI and CS. The Action requires entry into LCO 3.0.3 for various combinations of inoperability which are consistent with the present required actions for the same various combinations. The actual requirements are not being changed.
- A6 The existing Applicability for Core Spray System (CSS) Operability (3.5.A.1), and Low Pressure Coolant Injection (LPCI) Operability (3.5.B.1), requires both systems to be Operable whenever irradiated fuel is in the vessel and prior to startup from a COLD CONDITION. The proposed change (LCO 3.5.1 Applicability) requires them to be Operable in Modes 1, 2 and 3. This change more clearly defines the conditions when CSS and LPCI are required to be Operable without changing the specific requirements which are currently located in individual specifications for each system. This change is administrative because the same requirements for Operability currently listed in specific specifications will be labelled APPLICABILITY and apply to the entire ISTS Section 3.5.1, ECCS-Operating. The 3.5.A.2, 3.5.B.2, and 3.5.B.7 Applicabilities are only cross references and have been deleted.
- A7 The clarifying information contained in the "*" footnote has been moved to the proposed Bases for SR 3.5.1.2. The intent of the surveillance is to assure that the proper flow paths will exist for ECCS operation. The Bases clarifies that a valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. As such, moving this clarifying statement to the Bases is an administrative change.
- A8 This requirement has been deleted since it only provides reference to another Specification, and does not provide any unique requirements. The format of the proposed BFN ISTS does not include providing "cross references."
- A9 Surveillance Requirements for MOV operability, and check valves that are required by the Inservice Testing (IST) Program, have been removed from individual Specifications. This change is considered administrative in nature since these requirements remain in the IST Program which is defined by proposed Specification 5.5.6.

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING**

- A10 The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow is tested at both the higher and lower operating ranges of the system. Since the reactor steam dome pressure must be ≥ 920 psig to perform SR 3.5.1.7 and ≥ 150 psig to perform SR 3.5.1.8, sufficient time is allowed after adequate pressure is achieved to perform these tests. This is clarified by a Note in both SRs that state the Surveillances are not required to be performed until 12 hours after the specified reactor steam dome pressure is reached. CTS 3.5.E.1 already contains the context of the Note for the low pressure flow rate test. This is also consistent with interpretation of the current technical specification requirement for the high pressure flow rate test which is currently not modified by a Note.
- A11 The existing Applicabilities for High Pressure Coolant Injection (HPCI) Operability (3.5.E.1) and ADS (3.5.G.1) require the systems to be Operable whenever irradiated fuel is in the vessel and reactor pressure is greater than 150 psig (105 psig for ADS), except in the COLD SHUTDOWN CONDITION. The proposed change (LCO 3.5.1 Applicability) requires HPCI and ADS to be Operable in Modes 1, 2 and 3, except when reactor steam dome pressure is < 150 psig. (Reference Justification L5 for the change in applicability from < 105 psig to < 150 psig for ADS.) This change more clearly defines the conditions when HPCI and ADS are required to be Operable without changing the specific requirements which are currently located in the individual specifications. This change is administrative because the same requirements for Operability currently listed in the specific specifications will be labeled APPLICABILITY and apply to the entire ISTS Section 3.5.1, ECCS-Operating. The 3.5.E.2, 3.5.G.2, and 3.5.G.3 Applicabilities are only cross references and have been deleted.
- A12 A finite Completion Time has been provided to verify RCIC OPERABILITY. The new time is immediately and is considered administrative since this is an acceptable interpretation of the time to perform the current requirement.
- A13 CTS 3.9.A.3.h (for Unit 1 and 2) and 3.9.A.3.g (for Unit 3) require 480 V reactor motor operated valve (RMOV) boards to be energized with motor-generator (MG) sets in service. CTS 3.9.B.13 and 14 (for Unit 1 and 2) and 11 and 12 (for Unit 3) provide Required Actions for when one or any two 480-V MG board sets become inoperable. There are two 480-V AC RMOV boards that contain MG sets in their feeder lines. The 480-V AC RMOV boards provide motive power to valves associated with the LPCI mode of the RHR system. The MG sets act as electrical isolators to prevent a



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING**

fault propagating between electrical divisions due to an automatic transfer. Having an MG set out of service reduces the assurance that full RHR (LPCI) capacity will be available when required, therefore, the unit can only operate in this condition for 7 days. Having two MG sets out of service can considerably reduce equipment availability; therefore, the unit must be placed in Cold Shutdown within 24 hours. The inability to provide power to the inboard injection valve and the recirculation pump discharge valve from either 4 kV board associated with an inoperable MG set would result in declaring the associated LPCI subsystems inoperable and entering the Actions required for LPCI. Since, the out of service times for LPCI and the MG sets are comparable, the deletion of the MG set actions is considered administrative.

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 Proposed Action H requires LCO 3.0.3 be entered immediately which requires the plant to be in MODE 2 in 7 hours and MODE 3 within 13 hours when multiple ECCS subsystems are inoperable. This change is more restrictive because it stipulates that the reactor shutdown be completed much earlier than would be required by the existing specifications (CTS 3.5.A.3, 3.5.B.4, 3.5.B.8, and 3.5.E.3). For CTS 3.5.G.2 it is slightly more restrictive since it requires the plant to be in MODE 2 in 7 hours where no action was required before. CTS require a shutdown to MODE 4 within 24 hours (except CTS 3.5.G.2 for ADS which also requires the plant be in MODE 3 in 12 hours) but does not stipulate how quickly MODE 3 must be reached. Reference Comment L12 which addresses the less restrictive change of being in MODE 3 in 13 hours versus 12 hours and MODE 4 (or < 150 psig which is outside the applicability for ADS and HPCI) in 37 hours rather than 24 hours.
- M2 Surveillance requirement SR 3.5.1.3 has been added to verify that ADS air supply header pressure is ≥ 90 psig. This is a new Surveillance Requirement which verifies that sufficient air pressure exists in the ADS accumulators/receivers for reliable operation of ADS. Since this is a new Surveillance Requirement, it is an added restriction to plant operations.
- M3 With the reactor pressure < 105 psig, CTS 3.5.B.2 allows the RHR System to be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING**

that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE. This appears to be an exception to CTS 3.5.A.2 & 3, which only allows one CSS loop (i.e., one loop with two pumps) to be inoperable for 7 days and an immediate shutdown if this cannot be met. The # Note for 3.5.B.1 allows LPCI to be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure < 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable. Proposed Specification 3.5.1 has a similar provision (Note to SR 3.5.1.2). Since the proposed Specification has no provision that would allow continued operation in MODE 3 with pressure <105 psig with two CS loops with one pump per loop OPERABLE, the proposed change is considered more restrictive.

- M4 An additional requirement is being added that requires the plant to be in MODE 3 within 12 hours. This change is more restrictive because it stipulates that the reactor shutdown be completed much earlier than would be required by the existing specifications (CTS 3.5.A.3, 3.5.B.4, 3.5.B.8, and 3.5.E.3). CTS require a shutdown to MODE 4 within 24 hours but does not stipulate how quickly MODE 3 must be reached. Reference Comment L2 which addresses the less restrictive change of being in MODE 4 (or < 150 psig for HPCI and ADS) in 36 hours rather than 24 hours.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA1 Not used.

LA2 The details relating to system design and purpose have been relocated to the Bases. The design features and system operation are also described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the FSAR will be controlled by the provisions of 10 CFR 50.59. ECCS system operability determinations are described in the Bases. SR 3.5.1.1 will ensure maintenance of filled discharge piping.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING

- LA3 Details of the methods of performing surveillance test requirements and routine system status monitoring have been relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.
- LA4 Any time the OPERABILITY of a system or component has been affected by repair, maintenance or replacement of a component, post maintenance testing is required to demonstrate OPERABILITY of the system or component. Therefore, explicit post maintenance Surveillance Requirements have been deleted from the Specifications. Also, proposed SR 3.0.1 and SR 3.0.4 require Surveillances to be current prior to declaring components operable.
- LA5 CTS 3.5.D/4.5.D, Equipment Area Coolers, are being relocated to plant procedures. Relocating requirements for the equipment area coolers does not preclude them from being maintained operable. They are required to be operable in order to support HPCI, RCIC, LPCI and CS system operability. If they become inoperable, the operability of the supported systems are required to be evaluated under the Safety Function Determination Program in Section 5.0 of the Technical Specifications. This change is consistent with NUREG-1433.
- LA6 CTS 3.5.E specifically states that HPCI Operability can be determined prior to startup by using an auxiliary steam supply in lieu of using reactor steam after reactor steam dome pressure reaches 150 psig. Details of the methods of performing this surveillance test requirement have been relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.
- LA7 CTS 4.5.H.1 requires the discharge piping of RHR (LPCI and Containment Spray) to be vented from the high point and water level determined every month and prior to testing of these systems. The specific requirement to vent prior to testing has been relocated to procedures. Changes to the procedures will be controlled by the licensee controlled programs.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING

"Specific"

- L1 The phrase "actual or," in reference to the automatic initiation signal, has been added to the surveillance requirement for verifying the ECCS subsystems/ADS actuate on an automatic initiation signal. This allows satisfactory automatic system initiations for other than surveillance purposes to be used to fulfill this requirement. Operability is adequately demonstrated in either case since the ECCS subsystems/ADS itself can not discriminate between "actual" or "simulated."
- L2 The time to reach MODE 4, Cold Shutdown (for LPCI and CS) and < 150 psig (for HPCI and ADS) has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added for LPCI, CS and HPCI (Reference Comment M4 above). These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.
- L3 A new Action (proposed ACTION D) is being added to LCO 3.5.1 for the condition of an inoperable HPCI System coincident with one inoperable low pressure ECCS injection/spray subsystem. The analysis summarized in the current SAFER/GESTR-LOCA analysis (NEDC-32484P, February 1996) demonstrates that adequate cooling is provided by the ADS system and the remaining operable low pressure injection/spray subsystems. However, the redundancy has been reduced such that another single failure may not maintain the ability to provide adequate core cooling. Therefore, an allowable outage time of 72 hours has been assigned to restore either the inoperable HPCI system or the inoperable low pressure injection/spray subsystem to operability. This change is consistent with NUREG-1433.
- L4 The allowable outage time for HPCI has been extended from 7 days to 14 days. Adequate core cooling can be provided by ADS and the low pressure ECCS subsystems. The 14 days is allowed only if all six ADS valves and the low pressure ECCS subsystems are operable. (The exception, LCO 3.5.1, Condition D, which allows operation for 72 hours with HPCI and one low pressure ECCS subsystem inoperable is addressed in Comment L3 above.) The 14 day Completion Time is based on the reliability study that evaluated the impact on ECCS availability (Memorandum from R. L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING**

LCOs for ECCS Components," December 1, 1975). Factors contributing to the acceptability of allowing continued operations for 14 days with HPCI inoperable include: the similar functions of HPCI and RCIC, and that the RCIC is capable of performing the HPCI function, although at a substantially lower capacity; the continued availability of the full complement of ADS valves and the ADS System's capability in response to a small break LOCA; and, the continued availability of the full complement of low pressure ECCS subsystems which, in conjunction with ADS, are capable of responding to a small break LOCA. This change is consistent with NUREG-1433.

- L5 The pressure at which ADS is required to be operable is increased to 150 psig to provide consistency of the operability requirements for HPCI and RCIC equipment. Small break loss of coolant accidents are not analyzed to occur at low pressures (i.e., between 105 and 150 psig). The ADS is required to operate to lower the pressure sufficiently so that the LPCI and CS systems can provide makeup to mitigate such accidents. Since these systems can begin to inject water into the reactor pressure vessel at pressures well above 150 psig, there is no safety significance in the ADS not being operable between 105 and 150 psig.
- L6 A new ACTION has been added (ACTION F), which allows an outage time of 72 hours when one ADS valve and a low pressure ECCS subsystem is inoperable. Currently, there is no allowed outage time when these two items are inoperable. The analysis summarized in the current SAFER/GESTR-LOCA analysis (NEDC-32484P, February 1996) demonstrates that adequate cooling is provided by the HPCI and the remaining operable low pressure injection/spray system. However, the redundancy has been reduced such that another single failure concurrent with a design basis LOCA could result in the minimum required ECCS equipment not being available. Therefore, an allowable outage time of 72 hours has been assigned to restore either the inoperable ADS valve or the inoperable low pressure injection/spray system. This change is consistent with NUREG-1433.
- L7 Current Technical Specifications only allow one LPCI pump to be inoperable. Proposed ACTION A allows two LPCI pumps, one per loop or two in one loop, to be inoperable for seven days. The BASES for ISTS 3.5.1 Required Action A.1 state that the 7 day allowed outage time is justified because in this condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. This justification is applicable for the LPCI function of RHR with one or two RHR (LPCI) pumps out of service as demonstrated by previous LOCA analyses performed for



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING**

BFN as well as the current SAFER/GESTR-LOCA analysis (NEDC-32484P, February 1996). Following postulated single failures, adequate core cooling can be provided by one loop of Core Spray (2 pumps) and two RHR (LPCI) pumps (either two pumps in one loop or one pump in two loops) in conjunction with HPCI and ADS. Therefore, this less restrictive change is acceptable based on the plant specific LOCA analysis performed for BFN.

- L8 This change proposes to add a Note to current Surveillance Requirement 4.6.D.4 (proposed Surveillance Requirement 3.5.1.12) which states, "Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test." This change allows the Applicability of the Specification to be entered for 12 hours without performing the Surveillance Requirement. This allows for sufficient conditions to exist and allow the plant to stabilize within these conditions prior to performing the Surveillance. The normal outcome of the performance of a Surveillance is the successful completion which proves Operability. This change represents a relaxation over existing requirements. This change is consistent with NUREG-1433.
- L9 Existing Surveillance Requirement 4.5.E.1.d requires verification that HPCI is capable of delivering at least 5000 gpm at normal reactor vessel operating pressure. The proposed surveillance, SR 3.5.1.7, requires verification of a minimum 5000 gpm HPCI flow rate with reactor pressure ≥ 920 psig and < 1010 psig. The HPCI performance test at high pressure is the second part of a two part test that verifies HPCI pump performance at the upper and lower end of the range of steam supply and pump discharge pressures in which the HPCI pump is expected to perform. Performance of the HPCI test at both ends of the expected operating pressure range confirms that the HPCI pump and turbine are functioning in accordance with design specifications. The ability of the HPCI pump to perform at normal reactor vessel operating pressure has already been demonstrated. A small decrease in the pressure to as low as 920 psig at which the performance to design specifications is verified will not affect the validity of the test to determine that the pump and turbine are still operating at the design specifications.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING

L10 Existing Surveillance Requirement 4.5.C.1.e requires verification that HPCI is capable of delivering at least 5000 gpm "at 150 psig reactor steam pressure." The proposed surveillance, SR 3.5.1.9, requires verification of a minimum 5000 gpm HPCI flow rate with reactor pressure at \leq 165 psig. This change is less restrictive because it could allow reactor operation at pressures up to 165 psig prior to performing the surveillance. Performance of HPCI pump testing draws steam from the reactor and could affect reactor pressure significantly. Therefore, HPCI pump testing must be performed when the Electro-Hydraulic Control (EHC) System for the main turbine is available and capable of regulating reactor pressure. Operating experience has demonstrated that reactor pressures as high as 165 psig may be required before the EHC system is capable of maintaining stable pressure during the performance of the HPCI test.

The HPCI performance test at low pressure is the first part of a two part test that verifies HPCI pump performance at the upper and lower end of the range of steam supply and pump discharge pressures in which the HPCI pump is expected to perform. Performance of the HPCI test at both ends of the expected operating pressure range confirms that the HPCI pump and turbine are functioning in accordance with design specifications. The ability of the HPCI pump to perform at the lowest required pressure of 150 psig has already been demonstrated. A small increase in the pressure at which the performance to design specifications is verified will not significantly delay or affect the validity of the test to determine that the pump and turbine are still operating at the design specifications.

L11 CTS 3.5.E.1 requires HPCI operability to be determined within 12 hours after reactor steam dome pressure reaches 150 psig from a COLD CONDITION. The proposed Note to SR 3.5.1.7 and 3.5.1.8 allows 12 hours to perform the test after reactor steam dome pressure and flow are adequate. This is based on the need to reach conditions appropriate for testing. The existing allowance to reach a given pressure only partially addresses the issue. This pressure can be attained, and with little or no steam flow, conditions would not be adequate to perform the test - potentially resulting in an undesired reactor depressurization. The proposed change recognizes the necessary conditions of steam flow and minimum pressure as well as a maximum pressure limitation and provides consistency of presentation of these conditions. The point in time during startup that testing would begin remains unchanged. The change simply changes when the 12 hour clock for performing the test

JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING

must begin and permits testing to be completed in a reasonable period of time.

- L12 Proposed Condition H provides direction for various interrelationships between HPCI and ADS, and Between LPCI and CS. The Action requires entry into LCO 3.0.3 for various combinations of inoperability which are consistent with the present required actions for the same various combinations (CTS 3.5.A.3, 3.5.B.4, 3.5.B.8, and 3.5.E.3). However, the time to reach MODE 4, Cold Shutdown (for LPCI and CS) and < 150 psig (for HPCI) has been extended from 24 hours to 37 hours and to reach MODE 3, Hot Shutdown (for ADS only) has been extended from 12 hours to 13 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 2 in 7 hours and MODE 3 (Hot Shutdown) within 13 hours has been added (Reference Comment M1 above). These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.
- L13 An alternate verification to ensure the LPCI cross tie between loops is isolated has been added for Unit 3. The addition of an alternate method of satisfying the surveillance requirement is considered less restrictive. Currently, the method used for all three units is to verify the LPCI cross tie is closed and power is removed from the valve operator. Unit 3 has a manual shutoff valve install between the cross tie for Loop I and Loop II. This verification ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other subsystem. Since the manual shutoff valve serves the same function as the power operated valve, the proposed change is considered acceptable.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.1 - ECCS - OPERATING

RELOCATED SPECIFICATIONS

- RI. Browns Ferry Nuclear Plant consists of three units. The pump suction and heat exchanger discharge lines of one loop of RHR in Unit 1 (Loop II) are cross-connected to the pump suction and heat exchanger of Unit 2. Unit 2 and 3 systems are cross-connected in a similar manner. Technical Specification requirements related to RHR cross-tie capability between units have been deleted. The standby coolant supply connection and RHR crossties are provided to maintain long-term reactor core and primary containment cooling capability irrespective of primary containment integrity or operability of the RHR System associated with a given unit. They provide added long-term redundancy to the other ECC Systems and are designed to accommodate certain situations which, although unlikely to occur, could jeopardize the functioning of these systems. Neither the RHR cross-tie nor the standby coolant supply capability is assumed to function for mitigation of any transient or accident analyzed in the FSAR. Therefore, the operability requirements and surveillances associated with the cross-connection capability have been relocated to the Technical Requirements Manual (TRM). Changes to the TRM will be controlled in accordance with 10 CFR 50.59.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

AUG 02 1989

~~3.5/A.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.5.A Core Spray System (CSS)

See Justification for Changes for BFN ISTS 3.5.1

- 2. If one CSS loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are OPERABLE.
- 3. If Specification 3.5.A.1 or Specification 3.5.A.2 cannot be met, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

Applicability {

When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one OPERABLE pump and associated diesel generator shall be OPERABLE, except with the reactor vessel head removed as specified in 3.5.A.5 or PRIOR TO STARTUP as specified in 3.5.A.1.

LCO 3.5.2 L2

See Justification for Changes for BFN ISTS 3.5.1

4.5.A Core Spray System (CSS)

4.5.A.1.d (Cont'd)

105 psi differential pressure between the reactor vessel and the primary containment.

- e. Check Valve Per Specification 1.0.MM
- f. Verify that Once/Month each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct* position.
- 2. No additional surveillance is required.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

See Justification for Changes for BFN ISTS 3.5.2



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.A Core Spray System (CSS)~~

* 5. When irradiated fuel is in the reactor vessel and the reactor vessel head is removed, core spray is not required to be OPERABLE provided the cavity is flooded, the fuel pool gates are open and the fuel pool water level is maintained above the low level alarm point, and provided one ~~RHR~~ pump and associated valves supplying the standby coolant supply are OPERABLE.

LCO Applicability

(R1)

(M1) Proposed ACTIONS →

(M3)

Proposed SR 3.5.2.4

(M4)

Proposed SR 3.5.2.5 for CSS →

* When work is in progress which has the potential to drain the vessel, manual initiation capability of either 1 CSS Loop or 1 RHR pump, with the capability of injecting water into the reactor vessel, and the associated diesel generator(s) are required.

(L1)

See Justification for Changes for BFN ISTS 3.8.2

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

8. No additional surveillance required.

See Justification for Changes for BFN 15TS 3.5.1

SR3.5.2.5

Applicability 9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be (A2) demonstrated to be OPERABLE per Specification 1.0.MM.

See Justification for Changes for BFN 15TS 3.8.2

LCO 3.5.2 L2

Note for SR 3.5.2.4

(A1)

LCO Applicability

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

~~10. No additional surveillance required.~~

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification for Changes for BFN 15TS 3.5.1



MAY 19 1994

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.H. Maintenance of Filled Discharge Pipe~~

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

See Justification for Changes for BFN ISTS 3.5.1 + 3.5.3

~~4.5.H. Maintenance of Filled Discharge Pipe~~

SR 3.5.2.3

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined. (LA1)
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service. (LA2)
3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.
4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded. (LA3)



3.7/4.7 CONTAINMENT SYSTEMS

See Justification
Changes for BFN
ISTS 3.6.2.1+2

Specification 3.5.2

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

a. Minimum water level =
-6.25" (differential pressure control >0 psid)
-7.25" (0 psid differential pressure control)

b. Maximum water level =
-1"

SR 3.5.2.1

Add Applic of LCO 3.5.2

M2

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

~~A. Primary Containment~~

~~1. Pressure Suppression Chamber~~

SR 3.5.2.1

a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

M2

12 hrs



MAR 09 1994

~~3.9/A.9 AUXILIARY ELECTRICAL SYSTEM~~~~LIMITING CONDITIONS FOR OPERATION~~~~SURVEILLANCE REQUIREMENTS~~

(A1)

3.9.C. Operation in Cold Shutdown

Whenever the reactor is in COLD SHUTDOWN CONDITION with irradiated fuel in the reactor, the availability of electric power shall be as specified in Section 3.9.A except as specified herein.

1. At least two units 1 and 2 diesel generators and their associated 4-kV shutdown boards shall be OPERABLE.
2. An additional source of power energized and capable of supplying power to the units 1 and 2 shutdown boards consisting of at least one of the following:
 - a. One of the offsite power sources specified in 3.9.A.1.c.
 - b. A third OPERABLE diesel generator.
3. At least one 480-V shutdown board for each unit must be OPERABLE.

4. One 480-V RMOV board mg set is required for each RMOV board (1D or 1E) required to support operation of the RHR system in accordance with 3.5.B.9.

4.9.C. Operation in Cold Shutdown

1. No additional surveillance is required.

See Justification for Changes for BFN ISTS Section 3.8

(A3)



UNIT 2

CURRENT
TECHNICAL
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AUG 02 1989

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.5.A Core Spray System (CSS)

4.5.A Core Spray System (CSS)

4.5.A.1.d (Cont'd)

105 psi differential pressure between the reactor vessel and the primary containment.

- e. Check Valve Per Specification 1.0.MM
- f. Verify that Once/Month each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct* position.
- 2. No additional surveillance is required.

See Justification for Changes for BFN 1STS 3.5.1

- 2. If one CSS loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are OPERABLE.
- 3. If Specification 3.5.A.1 or Specification 3.5.A.2 cannot be met, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one OPERABLE pump and associated diesel generator shall be OPERABLE, except with the reactor vessel head removed as specified in 3.5.A.5 or PRIOR TO STARTUP as specified in 3.5.A.1.

Applicability LCO 3.5.2 L2

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

See Justification for Changes for BFN 1STS 3.5.1

See Justification for Changes for BFN 1STS 3.8.2



3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

DEC 15 1988

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.5.A ~~Core Spray System (CSS)~~

LCO 3.5.2
Applicability

- * 5. When irradiated fuel is in the reactor vessel and the reactor vessel head is removed, core spray is not required to be OPERABLE provided the cavity is flooded, the fuel pool gates are open and the fuel pool water level is maintained above the low level alarm point, and provided one RHR pump and associated valves supplying the standby coolant supply are OPERABLE.

(RI)

(MI) Proposed ACTIONS.

* When work is in progress which has the potential to drain the vessel, manual initiation capability of either 1 CSS Loop or 1 RHR pump, with the capability of injecting water into the reactor vessel, and the associated diesel generator(s) are required.

See Justification for Changes for BFN ISTS 3.8.2

(M3)
Proposed SR 3.5.2.4

(M4)
Proposed SR 3.5.2.5 for CSS

(LI)



~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

AI

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

8. No additional surveillance required.

See Justification for Changes for BFN ISTS 3.5.1

9. (When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.)

SR 3.5.2.5

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

See Justification for Changes for BFN ISTS 3.8.2

Applicability

LCO 3.5.2

L2

Note for SR 3.5.2.4

LCO Applicability

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

AI

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification for Changes for BFN ISTS 3.5.1



AUG 02 1989

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~~~LIMITING CONDITIONS FOR OPERATION~~

A1

~~SURVEILLANCE REQUIREMENTS~~~~3.5.H. Maintenance of Filled Discharge Pipe~~

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

See Justification for Change
for BFN ISTS 3.5.1 & 3.5.3

~~4.5.H. Maintenance of Filled Discharge Pipe~~~~SR 3.5.2.3~~

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined. (LA1)
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service. (LA2)
3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.
4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded. (LA3)

3.5/4.5-17

AMENDMENT NO. 169



See Justification for
Changes for BFN
1573 3.6.2.1+2

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

a. Minimum water level =
-6.25" (differential pressure control >0 psid)
-7.25" (0 psid differential pressure control)

b. Maximum water level =
-1"

SRs
3.5.2.1

Add applic of LCO 3.5.2

M2

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

1. Pressure Suppression Chamber

SR
3.5.2.1

a. The suppression chamber water level be checked once per ^{12hrs} day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

M2

JAN 09 1991

3.9/A.9 AUXILIARY ELECTRICAL SYSTEM

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.9.C. Operation in Cold Shutdown

Whenever the reactor is in COLD SHUTDOWN CONDITION with irradiated fuel in the reactor, the availability of electric power shall be as specified in Section 3.9.A except as specified herein.

1. At least two Units 1 and 2 diesel generators and their associated 4-kV shutdown boards shall be OPERABLE.
2. An additional source of power energized and capable of supplying power to the Units 1 and 2 shutdown boards consisting of at least one of the following:
 - a. One of the offsite power sources specified in 3.9.A.1.c.
 - b. A third OPERABLE diesel generator.
3. At least one 480-V shutdown board for each unit must be OPERABLE.

4. ~~One 480-V RMOV board mg set is required for each RMOV board (2D or 2E) required to support operation of the RHR system in accordance with 3.5.B.9.~~

4.9.C Operation in Cold Shutdown

- i. No additional surveillance is required.

See Justification for Changes for BFN ISTS Section 3.8

(A3)



UNIT 3
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

~~LIMITING CONDITIONS FOR OPERATION~~ (A1) ~~SURVEILLANCE REQUIREMENTS~~

3.5.A Core Spray System (CSS)

4.5.A Core Spray System (CSS)

4.5.A.1.d (Cont'd)

105 psi differential pressure between the reactor vessel and the primary containment.

e. Testable Check Valve Per Specification I.O.MM

f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month

2. No additional surveillance is required.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

- 2. If one CSS loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are OPERABLE.
- 3. If Specification 3.5.A.1 or Specification 3.5.A.2 cannot be met, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one OPERABLE pump and associated diesel generator shall be OPERABLE, except with the reactor vessel head removed as specified in 3.5.A.5 or PRIOR TO STARTUP as specified in 3.5.A.1.

Applicability {

LCO 3.5.2 {

(L2)

See Justification for Changes for BFN ISTS 3.5.1

See Justification for Changes for ISTS 3.5.1

See Justification for Changes for BFN ISTS 3.5.2



DEC 15 1988

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATIONS~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.A Core Spray System (CSS)~~

* 5. When irradiated fuel is in the reactor vessel and the reactor vessel head is removed, core spray is not required to be OPERABLE provided the cavity is flooded, the fuel pool gates are open and the fuel pool water level is maintained above the low level alarm point, and provided one RHRW pump and associated valves supplying the standby coolant supply are OPERABLE.

LCO 3.5.2
Applicability

(R1)

(M1)

Proposed ACTIONS

* When work is in progress which has the potential to drain the vessel, manual initiation capability of either 1 CSS Loop or 1 RHR pump, with the capability of injecting water into the reactor vessel, and the associated diesel generator(s) are required.

See Justification for Changes for BFN 1STS 3.8.2

(M3)

Proposed SR 3.5.2.4

(M4)

Proposed SR 3.5.2.5 for CSS

(L1)



Specification 3.5.2

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

APR 19 1994

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

8. No additional surveillance required.

See Justification for Changes for BFN 15TS 3.5.1

Applicability

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

SR 3.5.2.5
9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

See Justification for Changes for BFN 15TS 3.8.2

LCO 3.5.2 (2)

Note for SR 3.5.2.4

(A1)

LCO Applicability

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The B and D RHR pumps on unit 2 which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification for Changes for BFN 15TS 3.5.1



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.H Maintenance of Filled Discharge Pipe~~

~~4.5.H Maintenance of Filled Discharge Pipe~~

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

See Justification for Changes for BFN ISTS 3.5.1 & 3.5.3

SR 3.5.2.3

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow determined. (LA1)
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service. (LA2)
3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.
4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded. (LA3)



3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

SR 3.5.2.1

Add applicability of LCO 3.5.2

M2

- a. Minimum water level =
 - 6.25" (differential pressure control >0 psid)
 - 7.25" (0 psid differential pressure control)

- b. Maximum water level = -1"

See Justification for Changes for BFN ISTS 3.6.2.1 + 2

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

- ~~1. Pressure Suppression Chamber~~

SR 3.5.2.1

M2

12hrs day

- a. The suppression chamber water level be checked once per 12hrs day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.



(A1)

3.9.C. Operation in COLD SHUTDOWN CONDITION

Whenever the reactor is in the COLD SHUTDOWN CONDITION with irradiated fuel in the reactor, the availability of electric power shall be as specified in Section 3.9.A except as specified herein.

1. At least two Unit 3 diesel generators and their associated 4-kV shutdown boards shall be OPERABLE.
2. An additional source of power energized and capable of supplying power to the Unit 3 shutdown boards consisting of at least one of the following:
 - a. One of the offsite power sources specified in 3.9.A.1.c.
 - b. A third OPERABLE diesel generator.
3. At least one Unit 3 480-V shutdown board must be OPERABLE.

~~4. One 480-V RMOV board motor generator (mg) set is required for each RMOV board (3D or 3E) required to support operation of the RHR system in accordance with 3.5.B.9.~~

4.9.C Operation in COLD SHUTDOWN CONDITION

1. No additional surveillance is required.

see Justification for Changes for BFN 15TS Section 3.8

(A3)

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.2 - ECCS - SHUTDOWN**

ADMINISTRATIVE CHANGES

A1¹ Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 Surveillance Requirements for MOV operability that are required by the Inservice Testing (IST) Program have been removed from individual Specifications. This change is considered administrative in nature since these requirements remain in the IST Program which is defined by proposed Specification 5.5.6.

A3 CTS 3.9.C.4 requires one 480 V reactor motor operated valve (RMOV) board motor-generator (MG) set for each RMOV board required to support the RHR System in accordance with CTS 3.5.B.9. The 480-V AC RMOV boards provide motive power to valves associated with the LPCI mode of the RHR system. The MG sets act as electrical isolators to prevent a fault propagating between electrical divisions due to an automatic transfer. The inability to provide power to the inboard injection valve and the recirculation pump discharge valve from either 4 kV board associated with an inoperable MG set would result in declaring the associated LPCI subsystems inoperable and entering the Actions required for LPCI. Therefore, the deletion of the operability requirement associated with the MG sets in CTS 3.9.C.4 is considered administrative.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.2 - ECCS - SHUTDOWN

TECHNICAL CHANGE - MORE RESTRICTIVE

- M1 Proposed ACTIONS A, B, C and D have been added to provide required actions be taken when LCO requirements can not be met. CTS 3.5.A.4 and 3.5.B.9 provide minimum requirements for ECCS subsystems when in MODE 4 and 5 (except with the spent fuel pool gates removed and water level \geq the low level alarm setpoint of the spent fuel pool) but no action if these requirements are not met. Therefore, technical specifications are violated when these requirements can not be met and the default to TS 1.0.C.1 requires no action since the plant is already in Cold Shutdown. While from a compliance standpoint the proposed ACTIONS are less restrictive, from an operational perspective they are more restrictive since actions are required where there were none before. Proposed ACTION A allows 4 hours to restore a subsystem when only one of the required subsystems is inoperable and then proposed ACTION B requires action be initiated to suspend operations with a potential for draining the reactor vessel (OPDRVs) immediately. The 4 hour Completion Time is considered acceptable based on engineering judgment that considers the remaining available subsystem and the low probability of a vessel draindown event during this period. With no required ECCS injection spray subsystems inoperable, proposed ACTION C requires action to be initiated immediately to suspend OPDRVs and at least one required subsystem be restored to OPERABLE status within 4 hours. If one subsystem can not be restored within four hours then Proposed ACTION D requires action be initiated immediately to restore secondary containment to OPERABLE status, to restore two standby gas treatment systems to OPERABLE status, and to restore isolation capability in each required secondary containment penetration flow path not isolated. These actions must be immediately initiated to minimize the probability of a vessel draindown and the subsequent potential for fission product release.
- M2 Proposed SR 3.5.2.1 has been added. SR 3.5.2.1 requires the suppression pool water be verified \geq a minimum level every 12 hours. CTS 3.7.A.1 (& 4.7.A.1.a) requires the suppression pool be verified \geq -6.25" with no differential pressure control once per day at any time irradiated fuel is in the reactor vessel, and the nuclear system is pressurized or work is being done which has the potential to drain the vessel. Therefore, proposed SR 3.5.2.1 is more restrictive since the frequency of performance has been increased from once per 24 hours to once per 12 hours. In addition, CTS only requires performance during atmospheric conditions when work is being done that has the potential to drain the vessel. Therefore, the proposed SR is more restrictive since it



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.2 - ECCS - SHUTDOWN

requires performance during MODES 4, and 5, except with the spent fuel storage pool gates removed and water greater than or equal to minimum level over the top of the reactor pressure vessel flange. The CTS requirement to check the maximum level during OPDRVs has not been included since Specification 3.5.2 concerns the ability to maintain reactor water level using the suppression pool as a source of water. However, this level check is required for proposed Specifications 3.6.2.1 and 3.6.2.2 as it relates to Containment Systems.

- M3 Proposed SR 3.5.2.4, which requires a verification every 31 days that ECCS injection/spray valves are in their correct position, has been added. This provides assurance that the proper flow paths will exist for ECCS operation. This is more restrictive since BFN currently only requires this check during MODES 1, 2 and 3.
- M4 An SR has been added to require a system flow rate test for the Core Spray System during atmospheric conditions. While CTS (4.5.B.9) requires flow rate testing of the RHR pumps during atmospheric conditions as well as during MODES 1, 2, and 3, it only requires CSS flow rate testing during MODES 1, 2, and 3. The addition of this requirement is more restrictive.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

- LA1 CTS 4.5.H.1 requires the discharge piping of RHR (LPCI and Containment Spray) to be vented from the high point and water level determined every month and prior to testing of these systems. The specific requirement to vent prior to testing has been relocated to procedures. Changes to the procedures will be controlled by the licensee controlled programs.
- LA2 Any time the OPERABILITY of a system or component has been affected by repair, maintenance or replacement of a component, post maintenance testing is required to demonstrate OPERABILITY of the system or component. Therefore, explicit post maintenance Surveillance Requirements have been deleted from the Specifications. Also, proposed SR 3.0.1 and SR 3.0.4 require Surveillances to be current prior to declaring components operable.

PAGE 3 OF 5



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.2 - ECCS - SHUTDOWN**

LA3 Details of the methods of performing surveillance test requirements and routine system status monitoring have been relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

L1 CTS 3.5.A.5 requires manual initiation capability of either 1 CSS Loop or 1 RHR pump with capability of injecting water into the reactor vessel when work is in progress which has the potential to drain the vessel. The proposed Specification would not require the CSS or RHR (LPCI and containment cooling mode) system to be operable since LCO 3.5.2 applicability does not apply when the fuel pool gates are open and the fuel pool water level is maintained above the low level alarm setpoint. Therefore, the deletion of this requirement is considered less restrictive. The deletion is acceptable since the coolant inventory represented by this water level is sufficient to allow operator action to terminate the inventory loss prior to fuel uncovering in case of an inadvertent draindown.

L2 The proposed LCO for ECCS-Shutdown is less restrictive since it only requires two low pressure ECCS subsystems to be OPERABLE. This can be fulfilled with any combination of RHR and CS subsystems. That is, two CS subsystems (a CS subsystem for Specification 3.5.2 consists of at least one pump in one loop), two RHR subsystems (RHR subsystem for Specification 3.5.2 consists of one pump in one loop), or one RHR subsystem and one CS subsystem OPERABLE. CTS 3.5.B.9 requires one RHR loop with two pumps or two RHR loops with one pump per loop to be OPERABLE. CTS 3.5.B.4 requires one CS loop with one pump per loop to be OPERABLE. Per CTS 3.5.A Bases the minimum requirement at atmospheric pressure is for one supply of makeup water to the core. Therefore, requiring two RHR pumps and one CS pump to be OPERABLE provides excess redundancy. In addition, since only one supply of makeup water is required, sufficient makeup water can be provided by two CS subsystems, two RHR subsystems, or one CS and one RHR subsystem. As such, the proposed Specification ensures redundancy by requiring any two low pressure ECCS subsystems to be OPERABLE.

PAGE 4 OF 5

JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.2 - ECCS - SHUTDOWN

RELOCATED SPECIFICATIONS

R1 Browns Ferry Nuclear Plant consists of three units. The pump suction and heat exchanger discharge lines of one loop of RHR in Unit 1 (Loop II) are cross-connected to the pump suction and heat exchanger of Unit 2. Unit 2 and 3 systems are cross-connected in a similar manner. Technical Specification requirements related to RHR cross-tie capability between units have been deleted. The standby coolant supply connection and RHR crossties are provided to maintain long-term reactor core and primary containment cooling capability irrespective of primary containment integrity or operability of the RHR System associated with a given unit. They provide added long-term redundancy to the other ECC Systems and are designed to accommodate certain situations which, although unlikely to occur, could jeopardize the functioning of these systems. Neither the RHR cross-tie nor the standby coolant supply capability is assumed to function for mitigation of any transient or accident analyzed in the FSAR. Therefore, the operability requirements and surveillances associated with the cross-connection capability have been relocated to the Technical Requirements Manual (TRM). Relocation to the TRM is in accordance with the "Application of Selection Criteria to BFN TS" and the NRC Final Policy Statement on Technical Specification Improvements. Refer to the application document discussion for additional information.

UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

(A1)

3.5.E High Pressure Coolant Injection System (HPCIS)

See Justification for Changes for BFN ISTS 3.5.1

2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS (LPCI), and RCIGS are OPERABLE.

3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

e. Flow Rate at Once/18 150 psig months

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

f. Verify that Once/Month each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

~~F. Reactor Core Isolation Cooling System (RCIGS)~~

LCO 3.5.3

Applicability

1. The RCIGS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

BFN Unit 1

Proposed Note for SR 3.5.34

3.5/4.5-14

(A1) ~~F. Reactor Core Isolation Cooling System (RCIGS)~~

1. RCIG Subsystem testing shall be performed as follows:

(Actual of L1)

SR 3.5.3.5

a. Simulated Auto- Once/18 matic Actuation months Test

(A3)

Proposed Note for SR 3.5.3.5

AMENDMENT NO. 180

PAGE 2 OF 4



NOV 24 1989

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.F Reactor Core Isolation Cooling System (RCICS)~~

~~4.5.F Reactor Core Isolation Cooling System (RCICS)~~

~~3.5.F.1 (Cont'd)~~

~~4.5.F.1 (Cont'd)~~

Proposed Note for SR 3.5.3.4

be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

SR 3.5.3.3 b. Pump OPERABILITY

A2 Per Specification 1.0.MM

A4 c. Motor-Operated Valve OPERABILITY Per Specification 1.0.MM

SR 3.5.3.3 d. Flow Rate at normal reactor vessel operating pressure Proposed Note for SR 3.5.3.3

SR 3.5.3.4 e. Flow Rate at 150 psig ≤ 165 Once/18 months

SR 3.5.3.3 SR 7.5.7.4 The RCIC pump shall deliver at least 600 gpm during each flow test.

SR 3.5.3.2 f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct position.

A1 ~~2. No additional surveillances are required.~~

* Except that an automatic valve capable of automatic return to its normal position when a signal is present may be in a position for another mode of operation.

2. If the RCICS is inoperable, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is OPERABLE during such time.

ACTION A

3. If Specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurized to less than 150 psig within 24 hours.

L3 24 36 or equal to A8

M1 Be in made 3 in 12 hrs



MAY 19 1994

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.H. Maintenance of Filled Discharge Pipe~~

~~4.5.H. Maintenance of Filled Discharge Pipe~~

SR 3.5.3.1

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled. (LA1)

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined.
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

SR 3.5.3.1

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis. (LA2)

4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

See Justification for Changes for BFN 1STS 3.5.1



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



FEB 07 1991

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

- e. Flow Rate at Once/18 months
150 psig

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

- f. Verify that Once/Month each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

- 2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

See Justification for changes for BFN ISTS 3.5.1

- 2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS(LPCI), and RCIGS are OPERABLE.

- 3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

~~F. Reactor Core Isolation Cooling System (RCIGS)~~

(A1) ~~F. Reactor Core Isolation Cooling System (RCIGS)~~

LCO 35.3

- 1. The RCIGS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

Applicability

- 1. RCIG Subsystem testing shall be performed as follows:

SR3.5.3.5

- a. ^(Actual on L1) Simulated Automatic Actuation Test Once/18 months

(A2) Proposed Note for SR 3.5.3.5

BFN Unit 2

3.5/4.5-14

AMENDMENT NO. 190

Proposed Note for SR 3.5.3.4



A1

NOV 24 1989

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.F. Reactor Core Isolation Cooling System (RCIGS)

3.5.F.1 (Cont'd)

Proposed Note for SR 3.5.3.4

be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

L6 and flow

L43

ACTION A

2. If the RCIGS is inoperable, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is OPERABLE during such time.

verified immediately

3. If Specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurized to less than 150 psig within 24 hours.

36 or equal to

4.5.F. Reactor Core Isolation Cooling System (RCIGS)

4.5.F.1 (Cont'd)

SR 3.5.3.3 b. Pump OPERABILITY

Per Specification 1.0.MM

c. Motor-Operated Valve OPERABILITY Per Specification 1.0.MM

SR 3.5.3.3 d. Flow Rate at normal reactor vessel operating pressure

92 days Once/3 months 920 to 1010 psi's

SR 3.5.3.4 e. Flow Rate at 150 psig

150 psig ≤ 165

Once/18 months

The RCIG pump shall deliver at least 600 gpm during each flow test.

SR 3.5.3.2 f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct position.

Once/Month 31 days

2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its normal position when a signal is present may be in a position for another mode of operation.



AUG 02 1989

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.H. Maintenance of Filled Discharge Pipe

(A1)

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

(LA1)

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

See Justification for Changes for BFN ISTS 3.5.1

4.5.H. Maintenance of Filled Discharge Pipe

SR 3.5.3.1

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined.
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

SR 3.5.3.1

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

(LA2)

4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



(A1)

3.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E High Pressure Coolant Injection System (HPCIS)

4.5.E.1 (Cont'd)

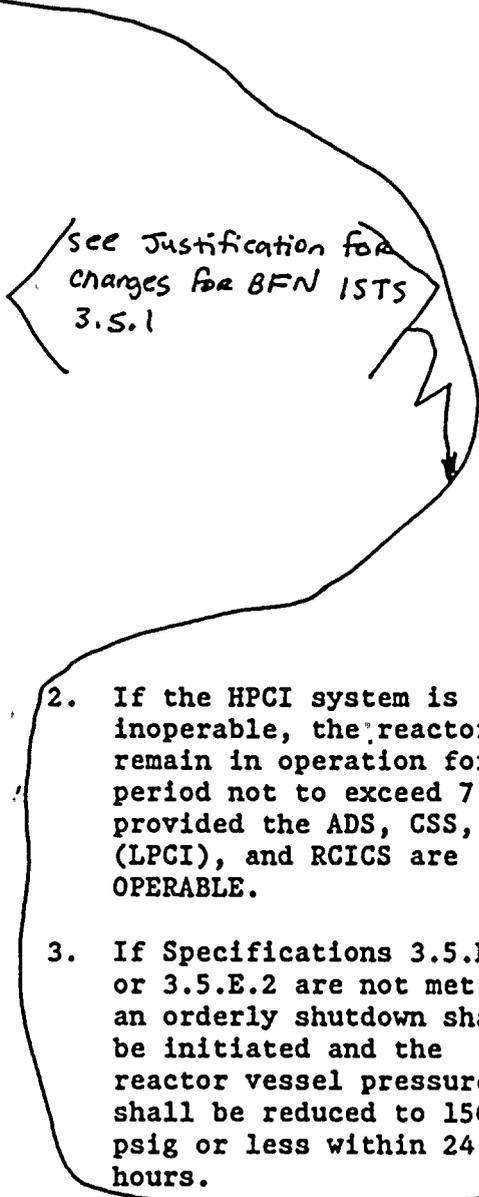
- e. Flow Rate at Once/18 months
150 psig

The HPCI pump shall deliver at least 5000 gpm during each flow rate test.

- f. Verify that Once/Month each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

- 2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



- 2. If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed 7 days, provided the ADS, CSS, RHRS (LPCI), and RCIGS are OPERABLE.
- 3. If Specifications 3.5.E.1 or 3.5.E.2 are not met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

~~F Reactor Core Isolation Cooling System (RCIGS)~~

~~F Reactor Core Isolation Cooling System (RCIGS)~~

LCO 3.5.3

Applicability

- 1. The RCIGS shall be OPERABLE whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is above 150 psig, except in the COLD SHUTDOWN CONDITION or as specified in 3.5.F.2. OPERABILITY shall

(A1)

- 1. RCIG Subsystem testing shall be performed as follows:

Actual or (L1)

- SR 3.5.3.5 a. Simulated Auto- Once/18 months Test

(A3) Proposed Note for SR 3.5.3.5

BFN Unit 3

Proposed Note for SR 3.5.3.4

3.5/4.5-14

AMENDMENT NO. 152



3.5.F Reactor Core Isolation Cooling System (RCIGS)

A1

3.5.F.1 (Cont'd)

L6 and flow

Proposed Note for SR 3.5.3.4

be determined within 12 hours after reactor steam pressure reaches 150 psig from a COLD CONDITION or alternatively PRIOR TO STARTUP by using an auxiliary steam supply.

LA3

2. If the RCIGS is inoperable, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is OPERABLE during such time. verified immediately

Action A

14 L2

3. If Specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurized to less than 150 psig within 24 hours. 36

L3

DR equal to AB

4.5.F Reactor Core Isolation Cooling System (RCIGS)

4.5.F.1 (Cont'd)

A2

SR 3.5.3.3 b. Pump OPERABILITY

Per Specification 1.0.MM

c. Motor-Operated Valve OPERABILITY

Per Specification 1.0.MM

SR 3.5.3.3 d. Flow Rate at normal reactor vessel operating pressure

Once/3 months

Proposed Note for SR 3.5.3.3

920 to 1010 psig

SR 3.5.3.4 e. Flow Rate at 150 psig

Once/18 months

≤ 165

The RCIG pump shall deliver at least 600 gpm during each flow test.

SR 3.5.3.2 f. Verify that each valve (manual, power-operated, or automatic) in the injection flowpath that is not locked, sealed, or otherwise secured in position, is in its correct position.

Once/Mon. 31 days

Be in Mode 3 in 12 hrs

2. No additional surveillances are required.

* Except that an automatic valve capable of automatic return to its normal position when a signal is present may be in a position for another mode of operation.



MAY 19 1994

~~2.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.H Maintenance of Filled Discharge Pipe~~

~~4.5.H Maintenance of Filled Discharge Pipe~~

(LAI) Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be OPERABLE, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

SR 3.5.3.1

The following surveillance requirements shall be adhered to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

1. Every month and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow determined.
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

SR 3.5.3.1.3

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

See Justification for Changes for BFN ISTS 3.5.1

4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.3 - RCIC SYSTEM

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The Frequency of "Once/month" has been changed to "31 days." The Frequencies of "Once/3 months" and "Per Specification 1.0MM" have been changed to "92 days." Since the proposed frequencies are equivalent, this change is considered administrative.
- A3 Notes allowing actual vessel injection to be excluded from this test (simulated automatic actuation test) have been added to proposed SR 3.5.3.5. Since the current requirements state the test is "simulated" (i.e., valve actuation and vessel injection are inherently excluded), this allowance is considered administrative in nature.
- A4 Surveillance Requirements for MOV operability that are required by the Inservice Testing Program have been removed from individual Specifications. This change is considered administrative in nature since these requirements remain in the IST Program which is defined by proposed Specification 5.5.6.
- A5 The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow is tested at both the higher and lower operating ranges of the system. Since the reactor steam dome pressure must be ≥ 920 psig to perform SR 3.5.3.3 and ≥ 150 psig to perform SR 3.5.3.4, sufficient time is allowed after adequate pressure is achieved to perform these tests. This is clarified by a Note in both SRs that state the Surveillances are not



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.3 - RCIC SYSTEM

required to be performed until 12 hours after the specified reactor steam dome pressure is reached. CTS 3.5.F.1 already contains the context of the Note for the low pressure flow rate test. This is also consistent with interpretation of the current technical specification requirement for the high pressure flow rate test which is currently not modified by a Note.

- A6 The clarifying information contained in the "*" footnote has been moved to the proposed Bases for SR 3.5.3.2. The intent of the surveillance is to assure that the proper flow paths will exist for RCIC System operation. The Bases clarifies that a valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. Moving this clarifying statement to the Bases is considered administrative in nature.
- A7 A finite Completion Time has been provided to verify HPCI OPERABILITY. the new time is immediately and is considered administrative since this is an acceptable interpretation of the time to perform the current requirement.
- A8 CTS 3.5.F.3 requires the reactor to be depressurized to less than 150 psig when CTS 3.5.F.1 and 2 cannot be met, while CTS 3.5.F.1 requires RCIC to be OPERABLE when reactor vessel pressure is above 150 psig. Proposed Required Action B.2 requires the vessel to be depressurized to \leq 150 psig. Since the intent of CTS is the same even though the CTS shutdown statement does not state "equal to," the addition of this requirement is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 An additional requirement is being added that requires the plant to be in MODE 3 within 12 hours. This change is more restrictive because it stipulates that the reactor shutdown be completed much earlier than would be required by the existing specification (CTS 3.5.F.3). CTS require a shutdown to \leq 150 psig within 24 hours but do not stipulate how quickly MODE 3 must be reached. Reference Comment L3 which addresses the less restrictive change of being \leq 150 psig in 36 hours rather than 24 hours.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.3 - RCIC SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 The details relating to system design and purpose have been relocated to the Bases. The design features and system operation are also described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the FSAR will be controlled by the provisions of 10 CFR 50.59. System operability determination, as described in the Bases and SR 3.5.3.1, will ensure maintenance of filled discharge piping.
- LA2 The details relating to methods of performing surveillance test requirements have been relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.
- LA3 CTS 3.5.F.1 specifically states that RCIC Operability can be determined prior to startup by using an auxiliary steam supply in lieu of using reactor steam after reactor steam dome pressure reaches 150 psig. Details of the methods of performing this surveillance test requirement have been relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 The phrase "actual or," in reference to the automatic initiation signal, has been added to the surveillance requirement for verifying that the RCIC System actuates on an automatic initiation signal. This allows satisfactory automatic system initiations for other than surveillance purposes to be used to fulfill the surveillance requirements. Operability is adequately demonstrated in either case since the RCIC System itself can not discriminate between "actual" or "simulated."
- L2 This change proposes to extend the current allowed outage time for the RCIC System from 7 days to 14 days. The 14 days are allowed only if the HPCI System is verified Operable immediately. Loss of the RCIC System will not affect the overall plant capability to provide makeup inventory at high reactor pressure since the HPCI System is the only high pressure system assumed to function during a LOCA. However, the RCIC System is



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.3 - RCIC SYSTEM

the preferred source of makeup for transients and certain abnormal events with no LOCA (RCIC as opposed to HPCI is the preferred source of makeup coolant because of its relatively small capacity, which allows easier control of the RPV water level). The 14 day completion time is also based on a reliability study that evaluated the impact on ECCS availability (Memorandum from R. L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975). Because of similar functions of HPCI and RCIC, and because HPCI is capable of performing the RCIC function, the allowed outage times determined for HPCI can be applied to RCIC. This change is consistent with NUREG-1433.

- L3 The time to reduce reactor steam dome pressure to ≤ 150 psig has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added (See Comment M1 above). These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.
- L4 Existing Surveillance Requirement 4.5.E.1.d requires verification that RCIC is capable of delivering at least 600 gpm at normal reactor vessel operating pressure. The proposed surveillance, SR 3.5.3.3, requires verification of a minimum 600 gpm RCIC flow rate with reactor pressure ≥ 920 psig and < 1010 psig. The RCIC performance test at high pressure is the second part of a two part test that verifies RCIC pump performance at the upper and lower end of the range of steam supply and pump discharge pressures in which the RCIC pump is expected to perform. Performance of the RCIC test at both ends of the expected operating pressure range confirms that the RCIC pump and turbine are functioning in accordance with design specifications. The ability of the RCIC pump to perform at normal reactor vessel operating pressure has already been demonstrated. A small decrease in the pressure to as low as 920 psig at which the performance to design specifications is verified will not affect the validity of the test to determine that the pump and turbine are still operating at the design specifications.
- L5 Existing Surveillance Requirement 4.5.F.1.e requires verification that RCIC is capable of delivering at least 600 gpm "at 150 psig reactor steam pressure." The proposed surveillance, SR 3.5.3.4, requires verification of a minimum 600 gpm RCIC flow rate with reactor pressure



JUSTIFICATION FOR CHANGES
BFN ISTS 3.5.3 - RCIC SYSTEM

at 165 psig. This change is less restrictive because it could allow reactor operation at pressures up to 165 psig prior to performing the surveillance. Performance of RCIC pump testing draws steam from the reactor and could affect reactor pressure significantly. Therefore, RCIC pump testing must be performed when the Electro-Hydraulic Control (EHC) System for the main turbine is available and capable of regulating reactor pressure. Operating experience has demonstrated that reactor pressures as high as 165 psig may be required before the EHC system is capable of maintaining stable pressure during the performance of the RCIC test.

The RCIC performance test at low pressure is the first part of a two part test that verifies RCIC pump performance at the upper and lower end of the range of steam supply and pump discharge pressures in which the RCIC pump is expected to perform. Performance of the RCIC test at both ends of the expected operating pressure range confirms that the RCIC pump and turbine are functioning in accordance with design specifications. The ability of the RCIC pump to perform at the lowest required pressure of 150 psig has already been demonstrated. A small increase in the pressure at which the performance to design specifications is verified will not significantly delay or affect the validity of the test to determine that the pump and turbine are still operating at the design specifications.

- L6 CTS 3.5.F.1 requires operability to be determined within 12 hours after reactor steam dome pressure reaches 150 psig from a COLD CONDITION. The allowance for reactor steam dome pressure and flow to be adequate is based on the need to reach conditions appropriate for testing. The existing allowance to reach a given pressure only partially addresses the issue. This pressure can be attained, and with little or no steam flow, conditions would not be adequate to perform the test - potentially resulting in an undesired reactor depressurization. The proposed change recognizes the necessary conditions of steam flow and minimum pressure as well as a maximum pressure limitation and provides consistency of presentation of these conditions. The point in time during startup that testing would begin remains unchanged. The change simply changes when the 12 hour clock for performing the test must begin and permits testing to be completed in a reasonable period of time.



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.5 - ECCS AND RCIC SYSTEM BASES**

The Bases of the current Technical Specifications for this section (3.5.A, B, E, F, G, H, and 4.5) have been completely replaced by revised Bases that reflect the format and applicable content of proposed BFN-UNIT 1, 2, and 3 ISTS Section 3.5, consistent with NUREG-1433. The revised Bases are as shown in the proposed BFN-UNIT 1, 2, and 3 Bases.

UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

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FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A. Primary Containment~~

(A2)

~~4.7.A. Primary Containment~~

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel ~~except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).~~

(OPERATE)

LCD 3.6.1.1 Applicability

(M1)

performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

(A2)

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

(A4)

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations), exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours

(LA1)

ACTION A { or the reactor shall be placed in Hot Shutdown
ACTION B { within the next 16 hours

(1)

(12)

(M2)

and cold shutdown in 36 hours

~~2. Integrated Leak Rate Testing~~

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

(LA1)

~~SR 3.6.1.1.1~~

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

~~3.7/4.7 CONTAINMENT SYSTEMS~~

FEB 22 1996

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~A.7.A. Primary Containment~~ c

~~A.7.A.2. (Cont'd)~~ c

(A1)

~~b. Deleted~~ c

~~c. Deleted~~ c

+
+

FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~ ←

~~4.7.A.2. (Cont'd)~~ ←

(A1)

~~d. Deleted~~ ←

+

~~e. Deleted~~ ←

+

~~f. Deleted~~ ←

+



(A1)

~~4.7.A. Primary Containment c~~

~~4.7.A.2. (Cont'd) c~~

SR 3.3.6.1.1

- g. Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN 15TS 3.6.1.2

Note: An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq (0.05 L_a)$ when tested at $\geq Pa$. (2) For door seal leakage, the overall air lock leakage rate is $\leq (0.02 L_a)$ when the air lock is pressurized to $(\geq 2.5 psig)$ for at least 15 minutes).

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(AG) →

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized, until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

ACTION A →

(1) (M2)

ACTION B →

in MODE 3 in 12 hours + MODE 4 in 36 hours

(M2)

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

See Justification for Changes for BFN ISTS 3.6.1.3

~~3.7/4.7 CONTAINMENT SYSTEMS~~

(A1)

~~INERTING CONNECTIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

J. Continuous Leak Rate Monitor

(LC1)

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

K. Drywell and Torus Surfaces

(A3)

The interior surfaces of the drywell and torus above the level one foot below the normal water line and outside surfaces of the torus below the water line shall be visually inspected each operating cycle for deterioration and any signs of structural damage with particular attention to piping connections and supports and for signs of distress or displacement.

~~3.7.4.7 CONTAINMENT SYSTEMS~~
~~OPERATING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

~~4.7.A Primary Containment~~

3.7.A.4 (Cont'd)

4.7.A.4 (Cont'd)

- c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.
- d. If Specifications 3.7.A.4.a, 3.7.A.4.b, or 3.7.A.4.c cannot be met, the unit shall be placed in a COLD SHUTDOWN CONDITION in an orderly manner within 24 hours.

- c. Each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.2.2.

- SR 3.6.1.1.2
- d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

See Justification for Changes for BFN 15TS 3.6.1.7

(A1)

(AS)

18 mos.

5. Oxygen Concentration

5. Oxygen Concentration

- a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.
- b. Within the 24-hour period subsequent to placing the reactor in the RUN MODE following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.
- c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a COLD SHUTDOWN CONDITION within 24 hours.
- d. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN CONDITION within 24 hours.

- a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.
- b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.
- c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

See Justification for Changes for BFN 15TS 3.6.3.2



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A. Primary Containment~~

~~4.7.A. Primary Containment~~

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

LCD 3.6.1.1
Applicability
M1

(A1)
(A2)
OPERABLE

performing "open vessel" physics tests at power levels not to exceed 5 MW(t).
(A2)

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

(A4)

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours

ACTION A For the reactor shall be placed in Hot Shutdown within the next 12 hours
ACTION B
1
M2

2. Integrated Leak Rate Testing
Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

(LAI)

SR 3.6.1.1.1
Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

And Cold Shutdown in 36 hours



~~3.7/4.7 CONTAINMENT SYSTEMS~~

(A1)

Specification 3.6.1.1

FEB 22 1996

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

~~b. Deleted~~

+

~~c. Deleted~~

+

FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

- (A1) ~~d. Deleted~~ †
- ~~e. Deleted~~ †
- ~~f. Deleted~~ †



AI

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

SR3 3.6.1.1

8. Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN ISTS 3.6.1.2

Note: An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq (0.05 L_a)$ when tested at \geq Pa. (2) For door seal leakage, the overall air lock leakage rate is $\leq (0.02 L_a)$ when the air lock is pressurized to ≥ 2.5 psig for at least 15 minutes).



~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

h: (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(A6) →

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

ACTION A →

(M2) →

ACTION B →

(M2)

in MODE 3 in 12 hours
+ MODE 4 in 36 hours

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

See Justification for Changes for BFN ISTS 3.6.1.3

(A1)

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

~~j. Continuous Leak Rate Monitor~~

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

(LC1)

~~k. Drywell and Torus Surfaces~~

The interior surfaces of the drywell and torus above the level one foot below the normal water line and outside surfaces of the torus below the water line shall be visually inspected each operating cycle for deterioration and any signs of structural damage with particular attention to piping connections and supports and for signs of distress or displacement.

(A3)



~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(AI)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

~~4.7.A Primary Containment~~

3.7.A.4 (Cont'd)

4.7.A.4 (Cont'd)

c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.

c. Each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.MM.

d. If Specifications 3.7.A.4.a, .b, or .c cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.

SR 3.6.1.1.2
d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

See Justification for changes for BFN ISTS 3.6.1.7

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.3.2

5. Oxygen Concentration

5. Oxygen Concentration

a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100/psig, except as specified in 3.7.A.5.b.

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a Cold Shutdown condition within 24 hours.

c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

d. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.



UNIT 3
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

FEB 22 1996

~~3.7/A.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

A1

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A. Primary Containment~~

A2

2.a. Primary containment integrity shall be **OPERABLE** maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

LCD 3.6.1.1
Applicability

M1

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

A4

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours

LAI

ACTION A { or the reactor shall be placed in Hot Shutdown
ACTION B { within the next 16 hours.

12

M2

and cold shutdown in 36 hours

~~4.7.A. Primary Containment~~

~~2. Integrated Leak Rate Testing~~

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.c shall be taken.

LAI

SR 3.6.1.1.1
Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

~~3.7/4.7 CONTAINMENT SYSTEMS~~

Specification 3.6.1.1

FEB 22 1996

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

(A1)

~~b. Deleted~~

+

~~c. Deleted~~

+

~~3.7/4.7 CONTAINMENT SYSTEMS~~ e

(A1)

Specification 3.6.1.1

FEB 22 1996

~~LIMITING CONDITIONS FOR OPERATION~~ e

~~SURVEILLANCE REQUIREMENTS~~ e

~~4.7.A. Primary Containment~~ e

~~4.7.A.2. (Cont'd)~~ e

(A1)

~~d. Deleted~~ e

+

~~e. Deleted~~ e

+

~~f. Deleted~~ e

+



~~4.7.A. Primary Containment e~~

~~4.7.A.2. (Cont'd) e~~

SR 3.3.6.1.1

8. Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

Note: An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq (0.05 L_a)$ when tested at \geq Pa. (2) For door seal leakage, the overall air lock leakage rate is $\leq (0.02 L_a)$ when the air lock is pressurized to ≥ 2.5 psig for at least 15 minutes).

See Justification for Changes for BFN ISTS 3.6.1.2

3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(A6)

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

ACTION A

(M2) 1

(M2)
in MODE 3 in 12 hours
+ MODE 4 in 36 hours

ACTION B

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

See Justification for Changes for BFN 15TS 3.6.1.3



(A1)

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

~~j. Continuous Leak Rate Monitoring~~

(LC1)

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

k. The interior surfaces of the drywell and torus above the level one foot below the normal water line and outside surfaces of the torus below the water line shall be visually inspected each operating cycle for deterioration and any signs of structural damage with particular attention to piping connections and supports and for signs of distress or displacement.

(A3)

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

~~4.7.A Primary Containment~~

3.7.A.4 (Cont'd)

4.7.A.4 (Cont'd)

c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.

c. Once each operating cycle, each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.MM.

d. If Specifications 3.7.A.4.a, 3.7.A.4.b, or 3.7.A.4.c, cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.

SR 3.6.1.1.2

d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

18 mos.

See Justification for changes for BFN ISTS 3.6.1.7

(A1)

(AS)

5. Oxygen Concentration

5. Oxygen Concentration

a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100/psig, except as specified in 3.7.A.5.b.

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a Cold Shutdown condition within 24 hours.

c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

d. If the specifications of 3.7.A.5.a through 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

See Justification for changes for BFN ISTS 3.6.3.2



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.1 - PRIMARY CONTAINMENT

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The definition of PRIMARY CONTAINMENT INTEGRITY has been deleted from the proposed Technical Specifications. In its place the requirement for primary containment is that it "shall be OPERABLE." This was done because of the confusion associated with these definitions compared to its use in the respective LCO. The change is editorial in that all the requirements are specifically addressed in the proposed LCO for the primary containment along with the remainder of the LCOs in the Containment Systems Primary Containment subsection (e.g., air locks, isolation valves, suppression pool). Therefore, the change is purely a presentation preference adopted by the BWR Standard Technical Specifications, NUREG 1433.

- A3 CTS 4.7.A.2.k requirements for visual inspection of the drywell and torus surfaces are also contained in 10 CFR 50, Appendix J. These regulations require licensee compliance and cannot be revised by the licensee. These details of the regulations within CTS are repetitious and unnecessary. Therefore, the details also found in Appendix J have been deleted. This is considered a presentation preference and as such is considered an administrative change.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.1 - PRIMARY CONTAINMENT

- A4 CTS 3.7.A.2.b provides acceptance criteria for integrated leak rate testing, which is redundant to those contained in Primary Containment Leakage Rate Testing Program (CTS 6.8.4.3) requirements. The definition of L_a is provided in proposed BFN ISTS 1.1 and need not be repeated here. As such, this deletion is considered administrative.
- A5 The acceptance criteria for the leak test of the drywell to suppression chamber structure has been changed from 0.09 lb/sec of primary containment atmosphere at 1 psid to 0.25 inches of water for 10 minutes. Since these values are equivalent this is considered an administrative change.
- A6 CTS 4.7.A.2.h(1) requires repairs to be initiated immediately when it is determined the criterion of 4.7.A.2.g is exceeded. CTS 4.7.A.2.g requires LLRTs to be performed in accordance with the Primary Containment Leakage Rate Testing Program (CTS 6.8.4.3). CTS 4.7.A.2.h(2) then allows 48 hours to demonstrate 4.7.A.2.g can be met following detection of excessive local leakage. Since repairs are typically initiated immediately and proposed BFN ISTS ACTION A will only allow 1 hour to restore primary containment to OPERABLE status prior to requiring the initiation of a shutdown (reference Justification M2 below), CTS 4.7.A.2.h(1) has been deleted.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 CTS 3.7.A.2.a requires the primary containment to be OPERABLE at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the vessel. The proposed BFN ISTS 3.6.1.1 applicability is MODES 1, 2, and 3. This is more restrictive since CTS does not require the primary containment to be OPERABLE when in MODE 2, not critical and < 212°F.
- M2 Proposed Action A is more restrictive than CTS 3.7.A.2.c since the time allowed to reduce excessive nitrogen leakage prior to initiating a shutdown has been reduced from 8 hours to 1 hour. The time allotted to place the unit in Hot Shutdown (MODE 3) has been reduced from 16 hours to 12 hours. Proposed Action B requires the unit to be placed in Cold Shutdown (MODE 4), whereas, CTS 3.7.A.2.c only requires the unit to be placed in Hot Shutdown.

In addition, CTS 4.7.A.2.h.(2) allows 48 hours to demonstrate conformance to Appendix J following detection of excessive local leakage



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.1 - PRIMARY CONTAINMENT

and then requires a plant shutdown if conformance can not be demonstrated. CTS does not specify a completion time for shutdown and does not specify whether shutdown is to the Hot or Cold Shutdown Condition. The Proposed Actions A and B are more restrictive since they only allow 1 hour to restore primary containment and then require the unit be in MODE 3 in 12 and MODE 4 in 36 hours.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 The details relating to routine monitoring of plant status and operations parameters that reflect primary containment operability and the methods of performing this monitoring have been relocated to the Bases and procedures. Acceptance criteria for primary containment N₂ leakage (i.e., makeup consumption) have been relocated to procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.
- LC1 The continuous leak rate monitor does not necessarily relate directly to primary containment operability. In general, the BWR Standard Technical Specifications, NUREG 1433, do not specify indication-only or alarm-only equipment to be OPERABLE to support operability of a system or component. Control of the availability of, and necessary compensatory activities if not available for, indications, monitoring instruments, and alarms are addressed by plant operational procedures and policies. Therefore, the continuous leak rate monitor, and associated alarm surveillances and actions will be relocated to a licensee controlled document. Any changes will require a 10 CFR 50.59 evaluation.

PAGE 3 OF 3



UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A. Primary Containment~~

~~4.7.A. Primary Containment~~

(A1)

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN ISTS 3.6.1.1

M4 - LCO 3.6.1.2 Applicability

Proposed ACTIONS A+B

M1 - Proposed Note 1+2 to ACTIONS

M3 - Proposed SR 3.6.1.1.2

FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

SR 3.6.1.2.1

8. Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN 1STS 3.6.1.1

Note: An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

SR 3.6.1.2.1
Note

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq (0.05 L_a)$ when tested at \geq Pa. (2) For door seal leakage, the overall air lock leakage rate is $\leq (0.02 L_a)$ when the air lock is pressurized to (≥ 2.5 psig for at least 15 minutes).

See Justification for Changes for BFN 1STS 5.5.12.

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(A2) →

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

Required Action C.2

Proposed Required Action C.1

ACTION D

in MODE 3 in 12 hours
+ MODE 4 in 36 hours

M2

24 M2

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

See Justification for Changes for BFN ISTS 3.6.1.3

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



FEB 22 1996

~~3.7/A.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A. Primary Containment~~

(A1) 2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

~~4.7.A. Primary Containment~~

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN ISTS 3.6.1.1

M4 - LCO 3.6.1.2 Applicability

M1 - Proposed ACTIONS A & B

Proposed Note 1 + 2 to ACTIONS

M3 - Proposed SR 3.6.1.1.2

FEB 22 1996

(A1)

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

SR 3.6.1.2.1

- 8. Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN ISTS 3.6.1.1

Note: An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

SR 3.6.1.2.1
Note

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq (0.05 L_a)$ when tested at \geq Pa. (2) For door seal leakage, the overall air lock leakage rate is $\leq (0.02 L_a)$ when the air lock is pressurized to $(\geq 2.5$ psig for at least 15 minutes).

See Justification for Changes for BFN ISTS 5.5.12

3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

A1

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

A2

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 15 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

Required Action C.2

Proposed Required Action C.1

ACTION D

in MODE 3 in 12 hours & MODE 4 in 36 hours

M2

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

See Justification for Changes for BFN ISTS 3.6.1.3



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A. Primary Containment~~

~~4.7.A. Primary Containment~~

(A1) 2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.c shall be taken.

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN ISTS 3.6.1.1

(M4) LCD 3.6.1.2 Applicability

Proposed ACTIONS A & B

(M1) Proposed Note 1 & 2 to Actions

(M3) Proposed SR 3.6.1.1.2



(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.A. Primary Containment~~

~~4.7.A.2. (Cont'd)~~

~~SR 3.6.1.2.1~~

8. Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Change for BFN ISTS 3.6.1.1

Note: An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

SR 3.6.1.2.1 Note

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq (0.05 L_a)$ when tested at \geq Pa. (2) For door seal leakage, the overall air lock leakage rate is $\leq (0.02 L_a)$ when the air lock is pressurized to (≥ 2.5 psig for at least 15 minutes).

See Justification for Change for BFN ISTS 5.5.12

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

A2

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

Required Action C.2

M2

Proposed Required Action C.1

ACTION D

in Mode 3 in 12 hrs or mode 4 in 36 hrs

M2

See Justification For Changes for BFN ISTS 3.6.1.3 in this section

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.2 - PRIMARY CONTAINMENT AIR LOCK

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 CTS 4.7.A.2.h(1) requires repairs to be initiated immediately when it is determined the criterion of 4.7.A.2.g is exceeded. CTS 4.7.A.2.g requires LLRTs to be performed in accordance with the Primary Containment Leakage Rate Testing Program (CTS 6.8.4.3). CTS 4.7.A.2.h(2) then allows 48 hours to demonstrate 4.7.A.2.g can be met following detection of excessive local leakage. Since repairs are typically initiated immediately and proposed Required Action C.1 for 3.6.1.2 requires action be initiated to evaluate the primary containment overall leakage rate using the current air lock results and ACTION A of ISTS 3.6.1.1 will only allow 1 hour to restore primary containment to OPERABLE status prior to requiring the initiation of a shutdown (reference Justification M2 for Specification 3.6.1.1), CTS 4.7.A.2.h(1) has been deleted.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 The current requirements for the air lock are located within the primary containment TS requirements. The current definition of primary containment integrity requires only one air lock door to be closed and sealed (i.e., the seal mechanism intact and sealing the door). Thus, no actions are required if one door is inoperable provided the other door is OPERABLE, since primary containment integrity only requires the one door. The proposed LCO requires the entire air lock to be OPERABLE,



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.2 - PRIMARY CONTAINMENT AIR LOCK

which includes both doors, as well as the interlock mechanism and the leak-tightness of the barrel. ACTIONS are provided (proposed ACTIONS A and B) to ensure that if one door or its interlock mechanism is inoperable, the other door is closed, locked and periodically verified to be closed and locked. If the interlock mechanism is inoperable, an allowance is provided to open the door provided a dedicated individual controls the access. Notes are provided to allow the locked closed verification to be performed administratively if the door is in a limited access area. These two new actions are not applicable, however, if the entire air lock is inoperable (as stated in proposed Note 1 to both ACTIONS A and B). To ensure that the primary containment LCO will be entered if air lock leakage results in exceeding overall primary containment leakage, NOTE 2 to the ACTIONS is also included. Overall, these new ACTIONS provide additional restrictions to plant operation.

- M2 CTS 4.7.A.2.h requires repairs to be initiated immediately when it is determined that the criterion of 4.7.A.2.g is exceeded and if conformance to these criterion is not demonstrated within 48 hours following detection of excessive local leakage, a reactor shutdown is required. ACTION C of the proposed Specification requires the licensee to initiate action to evaluate primary containment overall leakage rate using the current air lock test results immediately, verify an air lock door closed within 1 hour and restore the air lock to OPERABLE status within 24 hours. If required ACTION C and the associate Completion Time is not met, the unit must be in MODE 3 in 12 hours and MODE 4 in 36 hours. This is more restrictive than current requirements.
- M3 This change adds a Surveillance to verify the interlock mechanism works properly (only one door can be opened at a time). This will ensure that one door is always closed which maintains containment integrity. The addition of new requirements represents a more restrictive change.
- M4 The current requirements for the air lock are located within the primary containment TS requirements (CTS 3.7.A.2.a), which requires the primary containment to be OPERABLE at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the vessel. The proposed BFN ISTS 3.6.1.2 applicability is MODES 1, 2, and 3. This is more restrictive since CTS does not require the primary containment to be OPERABLE when in MODE 2, not critical, and < 212°F.

UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~~~SURVEILLANCE REQUIREMENTS~~~~3.7.A. Primary Containment~~

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

LCO 3.6.1.3
Applicability

M2

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for changes
for BFN 15-TS 3.6.1.1



(A1)

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

See Justification for Changes for BFN ISTS 3.6.1.1+3.6.1.2

SR 3.6.1.3.10

1. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling

(A2) outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.



NOV 16 1992

A1

3.7.C. Secondary Containment

- 3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
- 4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

See Justification for Changes for BFN 1STs 3.6.4.1

A1

except Reactor Building Vacuum breakers

~~D. Primary Containment Isolation Valves~~

M2

Applicability

- 1. When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE except as specified in 3.7.D.2.

LCD 3.6.1.3

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

L2

Note 1 to ACTIONS
Note 2 to SR 3.6.1.3.2

~~D. Primary Containment Isolation Valves~~

A1

- 1. The primary containment isolation valves surveillance shall be performed as follows:
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation

A2

SR 3.6.1.3.5
SR 3.6.1.3.6
SR 3.6.1.3.7

Actual or

L1

BFN Unit 1

3.7/4.7-17

AMENDMENT NO. 189



NOV 16 1992

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

~~3.7.D. Primary Containment Isolation Valves~~

~~4.7.D. Primary Containment Isolation Valves~~

~~4.7.D.1.a (Cont'd)~~

(A1)

SR 3.6.1.3.5

and in accordance with Specification 1.0.MM, tested for closure times.

b. In accordance with Specification 1.0.MM, all normally open power operated primary containment isolation valves shall be functionally tested.

c. (Deleted)

SR 3.6.1.3.8

d. At least once per operating cycle the OPERABILITY of the reactor coolant system instrument line flow check valves shall be verified.

EFCV actuates to the isolation position on a simulated instrument line break signal

(L5) Proposed ACTION B

(L9) Proposed ACTION D

(M2) Proposed ACTION F

(A3) Proposed Note 2 to ACTIONS

(A4) Proposed Notes 3+4 to ACTIONS

2. In the event any primary containment isolation valve becomes inoperable, reactor operation may continue provided at least one valve, in each line having an inoperable valve, is OPERABLE and within 4 hours either:

(A5)

Condition A+C

- a. The inoperable valve is restored to OPERABLE status, or
- b. Each affected line is isolated by use of at least one deactivated containment isolation valve secured in the isolated position.

Required ACTION A1+C1

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

ACTION E

(L8)

BFN Unit 1

The HOT SHUTDOWN CONDITION in 12 hours and in

(M1)

3.7/4.7-18

2. Whenever a primary containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

Required ACTION A2+C2

- Proposed SRs 3.6.1.3.1
- 3.6.1.3.2
- 3.6.1.3.3
- 3.6.1.3.4
- 3.6.1.3.9

a closed manual valve, blind flange, or check valve with flow through the valve secured

(L3)

AMENDMENT NO. 189



(A1)

FEB 13 1995

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.7.F. Primary Containment Purge System

1. The primary containment purge system shall be OPERABLE for PURGING, except as specified in 3.7.F.2.
 - a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.
 - b. The results of laboratory carbon sample analysis shall show $\geq 85\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.
 - c. System flow rate shall be shown to be within $\pm 10\%$ of design flow when tested in accordance with ANSI N510-1975.
2. If the provisions of 3.7.F.1.a, b, and c cannot be met, the system shall be declared inoperable. The provisions of Technical Specification 1.C.1 do not apply. PURGING may continue using the Standby Gas Treatment System.

3. a. The 18-inch primary containment isolation valves associated with PURGING may be open during the RUN mode for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. The OPERABILITY of

SR 3.6.1.3.1
Note

(L7)

LCo 3.6.1.3

4.7.F. Primary Containment Purge System

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8.5 inches of water at system design flow rate ($\pm 10\%$).
 - a. The tests and sample analysis of Specification 3.7.F.1 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first or after 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.
 - b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
 - c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

see Justification for Changes for LTS 3.7.F/4.7.F in this Section

AMENDMENT NO. 215



APR 29 1991

3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

(A)

SURVEILLANCE REQUIREMENTS

3.7.F. Primary Containment Purge System

4.7.F. Primary Containment Purge System

3.7.F.3 (Continued)

LCD 3.6.1.3 these primary containment isolation valves is governed by Technical Specification 3.7.D.

b. Pressure control of the containment is normally performed by VENTING through 2-inch primary containment isolation valves which route effluent to the Standby Gas Treatment System. The OPERABILITY of these primary containment isolation valves is governed by Technical Specification 3.7.D.

(LAI)

3.7.G. Containment Atmosphere Dilution System (CAD)

4.7.G. Containment Atmosphere Dilution System (CAD)

1. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE with:

1. System Operability

a. Two independent systems capable of supplying nitrogen to the drywell and torus.

a. Cycle each solenoid operated air/nitrogen valve through at least one complete cycle of full travel in accordance with Specification 1.0.MM, and at least once per month verify that each manual valve in the flow path is open.

b. A minimum supply of 2,500 gallons of liquid nitrogen per system.

b. Verify that the CAD System contains a minimum supply of 2,500 gallons of liquid nitrogen twice per week.

See Justification for Change to BFN 15TS 3.6.3.1

3.7/4.7-22

AMENDMENT NO. 184

INSERT PROPOSED NEW SPECIFICATION 3.6.1.4

Insert new Specification 3.6.1.4, "Drywell Air Temperature," as shown in the BFN Unit 2 Improved Standard Technical Specifications.



JUSTIFICATION FOR CHANGES
BFN ISTS: 3.6.1.4 - DRYWELL AIR TEMPERATURE

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 A new Specification is being added requiring drywell air temperature to be $\leq 150^{\circ}\text{F}$. This is required since some accident analyses assume this temperature at the start of an accident. Appropriate ACTIONS and Surveillance Requirements are also added. This is consistent with the BWR Standard Technical Specifications, NUREG 1433.



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

FEB 22 1996

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A. Primary Containment~~

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or, when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while

LCD 3.6.1.3
Applicability

(M2)

performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN 1STS 3.6.1.1

(A1)

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

See Justification for Changes for BFN ISTS 3.6.1.1 + 3.6.1.2

SR 3.6.1.3.10

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling

(A2) outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.



NOV 16 1992

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.7.C. Secondary Containment

- 3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
- 4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.4.1

(A1)

D. Primary Containment Isolation Valves

- 1. (When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE except as specified in 3.7.D.2.

(M2) Applicability

LCO 3.6.1.3

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

(L2)

Note 1 to ACTIONS
Note 2 to SR 3.6.1.3.2

BFN Unit 2

3.7/4.7-17

(A1)

except Reactor Building vacuum breakers

(A1)

D. Primary Containment Isolation Valves

- 1. The primary containment isolation valves surveillance shall be performed as follows:
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation

(A2)

SR 3.6.1.3.5
SR 3.6.1.3.6
SR 3.6.1.3.7

actual or (L1)

AMENDMENT NO. 204



~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.D. Primary Containment Isolation Valves~~

~~4.7.D. Primary Containment Isolation Valves~~

~~4.7.D.1.a (Cont'd)~~ (A1)

SR 3.6.1.3.5
SR 3.6.1.3.6

and in accordance with Specification 1.0.MM, tested for closure times.

b. In accordance with Specification 1.0.MM, all normally open power operated primary containment isolation valves shall be functionally tested.

c. (Deleted)

SR 3.6.1.3.8

d. At least once per operating cycle the OPERABILITY of the reactor coolant system instrument line flow check valves shall be verified. (A2)

(M4)

EFCV actuates to the isolation position on a simulated instrument line break signal

(L5) Proposed ACTION B

(L9) Proposed ACTION D

(M2) Proposed ACTION F

(A3) Proposed Note 2 to ACTIONS

(A4) Proposed Notes 3+4 to ACTIONS

2. In the event any primary containment isolation valve becomes inoperable, reactor operation may continue provided at least one valve, in each line having an inoperable valve, is OPERABLE and within 4 hours either:

(A5)

Condition A+C

a. The inoperable valve is restored to OPERABLE status, or

b. Each affected line is isolated by use of at least one deactivated containment isolation valve secured in the isolated position.

Required Action A.1+C.1

(L4)

2. Whenever a primary containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily. (L6)

Required Action A2+C.2

Proposed SRs 3.6.1.3.1, 3.6.1.3.2, 3.6.1.3.3, 3.6.1.3.4, 3.6.1.3.9 (M3)

ACTIONS A+C

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

ACTION E

(L8) 36

BFN Unit 2

the HOT SHUTDOWN CONDITION in 12 hours and in (M1)

3.7/4.7-18

AMENDMENT NO. 204

PAGE 5 OF 7

FEB 13 1995

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.7.F. Primary Containment Purge System

- 1. The primary containment purge system shall be OPERABLE for PURGING, except as specified in 3.7.F.2.
 - a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.
 - b. The results of laboratory carbon sample analysis shall show $\geq 85\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.
 - c. System flow rate shall be shown to be within $\pm 10\%$ of design flow when tested in accordance with ANSI N510-1975.
- 2. If the provisions of 3.7.F.1.a, b, and c cannot be met, the system shall be declared inoperable. The provisions of Technical Specification 1.C.1 do not apply. PURGING may continue using the Standby Gas Treatment System.

T

SR3.6.1.3.1 Note

(L7)

3. a. The 18-inch primary containment isolation valves associated with PURGING may be open during the RUN mode for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. The OPERABILITY of

4.7.F. Primary Containment Purge System

- 1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8.5 inches of water at system design flow rate ($\pm 10\%$).
 - a. The tests and sample analysis of Specification 3.7.F.1 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first or after 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.
 - b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
 - c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

SEE JUSTIFICATION FOR CHANGES FOR CTS 3.7.F/4.7.F in 17; Section

AMENDMENT NO. 231



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.F. Primary Containment Purge System~~

(AI)

~~3.7.F.3 (Continued)~~

LCo 3.6.1.3 these primary containment isolation valves is governed by Technical Specification 3.7.D.

- b. Pressure control of the containment is normally performed by VENTING through 2-inch primary containment isolation valves which route effluent to the Standby Gas Treatment System. The OPERABILITY of these primary containment isolation valves is governed by Technical Specification 3.7.D.

~~4.7.F. Primary Containment Purge System~~

(LAI)

~~3.7.G. Containment Atmosphere Dilution System (CAD)~~

1. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE with:

- a. Two independent systems capable of supplying nitrogen to the drywell and torus.

- b. A minimum supply of 2,500 gallons of liquid nitrogen per system.

~~4.7.G. Containment Atmosphere Dilution System (CAD)~~

1. System Operability

- a. Cycle each solenoid operated air/nitrogen valve through at least one complete cycle of full travel in accordance with Specification 1.0.MM, and at least once per month verify that each manual valve in the flow path is open.

- b. Verify that the CAD System contains a minimum supply of 2,500 gallons of liquid nitrogen twice per week.

BFN Unit 2

SEE JUSTIFICATION FOR CHANGE FOR BFN ISTS 3.6.3, 3.7/4.7-22

AMENDMENT NO. 197



INSERT PROPOSED NEW SPECIFICATION 3.6.1.4

Insert new Specification 3.6.1.4, "Drywell Air Temperature," as shown in the BFN Unit 2 Improved Standard Technical Specifications.

JUSTIFICATION FOR CHANGES
BFN ISTS: 3.6.1.4 - DRYWELL AIR TEMPERATURE

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 A new Specification is being added requiring drywell air temperature to be $\leq 150^{\circ}\text{F}$. This is required since some accident analyses assume this temperature at the start of an accident. Appropriate ACTIONS and Surveillance Requirements are also added. This is consistent with the BWR Standard Technical Specifications, NUREG 1433.



UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

(A1)

~~3.7.A. Primary Containment~~

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

LCD 3.6.1.3
Applicability
Th2

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a , does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a .

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.c shall be taken.

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

See Justification for Changes for BFN 1STS 3.6.1.1



~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~MITTING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

See Justification for Changes for BFN ISTS 3.6.1.1 & 3.6.1.2

SR 3.6.1.3.10

1. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

(A2)



(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.7.C. Secondary Containment

- 3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
- 4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

See Justification for Changes for BFN ISTS 3.6.4.1

(A1)

~~D. Primary Containment Isolation Valves~~

m2
Applicability

- 1. When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE except as specified in 3.7.D.2.

LCO 3.6.1.3

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

(L2)

Note 1 to ACTIONS
Note 2 to SR 3.6.1.3.2

(A1)

except Reactor Building Vacuum breakers

(A1)

~~D. Primary Containment Isolation Valves~~

- 1. The primary containment isolation valves surveillance shall be performed as follows: (A2)
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation

SR 3.6.1.3.5
SR 3.6.1.3.6
SR 3.6.1.3.7

actual OR

(L1)



~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.D. Primary Containment Isolation Valves~~

~~4.7.D. Primary Containment Isolation Valves~~

(A1)

~~4.7.D.1.a (Cont'd)~~

(L5) Proposed Action B

(L9) Proposed Action D

(M2) Proposed Action F

(A3) Proposed Note 2 to Actions

(A4) Proposed Notes 3+4 to Actions

SR 3.6.1.3.5 and in accordance with Specification 1.0.MM, tested for closure times.

b. In accordance with Specification 1.0.MM, all normally open power operated primary containment isolation valves shall be functionally tested

c. (Deleted)

SR 3.6.1.3.8

d. At least once per operating cycle the OPERABILITY of the reactor coolant system instrument line flow check valves shall be verified.

(A2)

(M4) EPCV actuates to the isolation position on a Simulated Instrument line Break signal

(A5) 2. In the event any primary containment isolation valve becomes inoperable, reactor operation may continue provided at least one valve, in each line having an inoperable valve, is OPERABLE and within 4 hours either:

Condition A+C

a. The inoperable valve is restored to OPERABLE status, or

b. Each affected line is isolated by use of at least one deactivated containment isolation valve secured in the isolated position.

Required Action A.1 + C.1

ACTIONS A+C

2. Whenever a primary containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

Required Action A.2 + C.2

Proposed SRs 3.6.1.3.1
3.6.1.3.2
3.6.1.3.3
3.6.1.3.4
3.6.1.3.9

(L3) a closed manual valve, blind Flange, or check valve with flow through the valve secured

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

Action E

(L8) (L9)

BFN Unit 3

the hot shutdown condition in 12 hours and in

3.7/4.7-18

(M1)

3.7.F. Primary Containment Purge System

4.7.F. Primary Containment Purge System

- 1. The primary containment purge system shall be OPERABLE for PURGING, except as specified in 3.7.F.2.
 - a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.
 - b. The results of laboratory carbon sample analysis shall show $\geq 85\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.
 - c. System flow rate shall be shown to be within $\pm 10\%$ of design flow when tested in accordance with ANSI N510-1975.
- 2. If the provisions of 3.7.F.1.a, b, and c cannot be met, the system shall be declared inoperable. The provisions of Technical Specification 1.C.1 do not apply. PURGING may continue using the Standby Gas Treatment System.

- 1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8.5 inches of water at system design flow rate ($\pm 10\%$).
 - a. The tests and sample analysis of Specification 3.7.F.1 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first or after 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.
 - b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
 - c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

- 3. a. The 18-inch primary containment isolation valves associated with PURGING may be open during the RUN mode for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. The OPERABILITY of

SR 3.6.1.3.1 Note

(L7)

See Justification for Changes for CTS 3.7.F/4.7.F in this Section

LCD 3.6.1.3

3.7/4.7-21

AMENDMENT NO. 188

PAGE 6 OF 7



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

~~3.7.F. Primary Containment Purge System~~

~~4.7.F. Primary Containment Purge System~~

~~3.7.F.3 (Continued)~~

LCO 3.6.1.3 these primary containment isolation valves is governed by Technical Specification 3.7.D.

b. Pressure control of the containment is normally performed by VENTING through 2-inch primary containment isolation valves which route effluent to the Standby Gas Treatment System. The OPERABILITY of these primary containment isolation valves is governed by Technical Specification 3.7.D.

LAI

3.7.G. Containment Atmosphere Dilution System (CAD)

4.7.G. Containment Atmosphere Dilution System (CAD)

1. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE with:

1. System Operability

a. Two independent systems capable of supplying nitrogen to the drywell and torus.

a. Cycle each solenoid operated air/nitrogen valve through at least one complete cycle of full travel in accordance with Specification 1.0.MM, and at least once per month verify that each manual valve in the flow path is open.

b. A minimum supply of 2,500 gallons of liquid nitrogen per system.

b. Verify that the CAD System contains a minimum supply of 2,500 gallons of liquid nitrogen twice per week.

See Justification for Changes for BFN 15TS 3.6.3.1

3.7/4.7-22

INSERT PROPOSED NEW SPECIFICATION 3.6.1.4

Insert new Specification 3.6.1.4, "Drywell Air Temperature," as shown in the BFN Unit 2 Improved Standard Technical Specifications.



JUSTIFICATION FOR CHANGES
BFN ISTS: 3.6.1.4 - DRYWELL AIR TEMPERATURE

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 A new Specification is being added requiring drywell air temperature to be $\leq 150^{\circ}\text{F}$. This is required since some accident analyses assume this temperature at the start of an accident. Appropriate ACTIONS and Surveillance Requirements are also added. This is consistent with the BWR Standard Technical Specifications, NUREG 1433.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

In addition, the PCIV LCO now exempts the reactor building-to-suppression chamber vacuum breakers and scram discharge volume vent and drain valves since they are governed by other LCOs. Any changes to the requirements for these valves are discussed in the new LCO Justification for Changes.

- A2 The current technical specification (CTS) 4.7.D.1.a frequency of "once per operating cycle" has been changed to "In accordance with the Inservice Testing Program" for proposed SR 3.6.1.3.5 (stroke time tests). The CTS 4.7.D.1.d frequency of "once per operating cycle" has been changed to "18 months" for proposed SR 3.6.1.3.8. Since an operating cycle is 18 months and the current IST program requires testing every 18 months, this change is considered administrative in nature. The CTS 4.7.A.2.i frequency of "each refueling outage" has been replaced with "in accordance with the Primary Containment Leakage Rate Testing Program" for SR 3.6.1.3.10. This program requires Appendix J requirements to be met. The Appendix J requirements will always supersede the Technical Specification requirements (unless an exemption is approved) since Appendix J is the rule. Therefore, this change is purely an administrative preference in presentation.
- A3 This proposed Note ("Separate Condition entry is allowed for each penetration flow path") provides explicit instructions for proper application of the actions for Technical Specification compliance. In



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

conjunction with the proposed Specification 1.3 - "Completion Times," this Note provides direction consistent with the intent of the existing Actions for inoperable isolation valves.

- A4 The proposed ACTIONS include Notes 3 and 4. These Notes facilitate the use and understanding of the intent to consider any system affected by inoperable isolation valves, which is to have its ACTIONS also apply if it is determined to be inoperable. Note 4 clarifies that these "systems" include the primary containment. With proposed LCO 3.0.6, this intent would not necessarily apply. This clarification is consistent with the intent and interpretation of the existing Technical Specifications, and is therefore considered an administrative presentation preference.
- A5 The current single Action for "any primary containment isolation valve" has been divided into three ACTIONS. Proposed ACTION A for one valve inoperable in a penetration that has two valves, proposed ACTION B for two valves inoperable in a penetration that has two valves, and proposed ACTION C for one valve inoperable in a penetration that has only one valve. All technical changes are discussed elsewhere in this section. As such, this change is considered an administrative presentation preference.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 CTS 3.7.D.3 requires an orderly shutdown be initiated and the reactor to be in the COLD SHUTDOWN CONDITION within 24 hours when certain conditions can not be met. Proposed Action E will require the plant be in MODE 3 in 12 hours and MODE 4 in 36 hours. The addition of this intermediate step to the COLD SHUTDOWN CONDITION is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.
- M2 CTS applicability for PCIV operability is when primary containment integrity is required. Per CTS 3.7.A.2.a, primary containment integrity is required at all times the reactor is critical or when the reactor water temperature is > 212°F and fuel is in the reactor vessel. The proposed applicability of MODES 1, 2, and 3 is more restrictive since CTS does not require primary containment integrity when in MODE 2, not

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

critical and < 212°F. The proposed Specification is also applicable when associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, which adds a MODE 4 and 5 requirement for the RHR Shutdown Cooling isolation valves. An appropriate ACTION has been added (proposed ACTION F) for when the valves cannot be isolated (since the unit is already in MODE 4 or 5, the current actions provide no appropriate compensatory measures). ACTION F requires the licensee to initiate action to suspend operations with the potential for draining the reactor vessel immediately and to restore valve(s) to OPERABLE status immediately. If suspending an OPDRV would result in closing the RHR Shutdown Cooling valves, an alternative required action is provided to immediately initiate action to restore the valves to OPERABLE status.

- M3 New Surveillance Requirements have been added. SRs 3.6.1.3.1, 3.6.1.3.2 and 3.6.1.3.3 ensure PCIVs are in their proper position or state. SRs 3.6.1.3.4 and 3.6.1.3.9 ensure the traversing incore probe (TIP) squib valves will actuate if required. These SRs are additional restrictions on plant operation.
- M4 This change adds acceptance criteria to the Surveillance Requirement which requires an Operability test of the instrument line excess flow check valves (EFCVs). The acceptance criteria added requires that the EFCVs actuate to the isolation position on a simulated instrument line break signal. The addition of acceptance criteria which did not previously exist in Technical Specifications constitutes a more restrictive change.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 CTS 3.7.F.3.b provides no requirements, it just explains that the normal method of containment pressure control is through 2-inch PCIVs, which route effluent through the SGTS. Since the OPERABILITY of these valves is governed by proposed BFN ISTS 3.6.1.3, the specification provides no requirements and has been eliminated. Any details relating to PCIV operability have been relocated to the Bases of LCO 3.6.1.3. Placing these details in the Bases provides assurance they will be appropriately maintained since changes to these details will require a 50.59 evaluation.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

"Specific"

- L1 The phrase "actual or" in reference to the automatic isolation signal, has been added to the Surveillance Requirement for verifying that each PCIV actuates on an automatic isolation signal. This allows satisfactory automatic PCIV isolations for other than Surveillance purposes to be used to fulfill the Surveillance Requirements. Operability is adequately demonstrated in either case since the PCIV cannot discriminate between "actual" or "simulated".
- L2 The provisions of the "*" Note of CTS 3.7.D.1 are encompassed by Note 1 to the ACTIONS, which allows penetration flow paths to be unisolated intermittently under administrative controls (except for the 18-inch purge valve penetration flow paths). However, the ISTS allowance applies to all primary containment isolation valves (except for 18-inch purge valve penetration flow paths) not just locked or sealed closed valves. The allowance is presented in proposed ACTIONS Note 1 and in SR 3.6.1.3.2, Note 2. Opening of primary containment penetrations on an intermittent basis is required for performing surveillances, repairs, routine evolutions, etc.
- L3 CTS 3.7.D.2.b allows isolating the primary containment penetrations with at least one deactivated valve secured in the isolated position when one PCIV is inoperable. The proposed ACTIONS A and C of LCO 3.6.1.3 allow the use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve (for Condition A only) with flow through the valve secured. The Action utilizing a deactivated automatic or manual valve is appropriate on the basis that these isolations present a boundary which is not affected by a single failure. The ability to utilize the valves downstream of the outboard PCIVs is an acceptable isolation since it meets the acceptance criteria of not being affected by a single active failure.
- L4 CTS 3.7.D.2 allows reactor operation to continue when any PCIV becomes inoperable provided that at least one valve in each line having an inoperable valve is operable and within 4 hours the affected line is isolated or the inoperable valve is restored to OPERABLE status. Based on the wording, this only applies to lines with two isolation valves. This is equivalent to proposed ACTION A, however, the proposed ACTION allows additional time to isolate the main steam lines. A Completion Time of 8 hours for the MSLs allows a period of time to restore the MSIVs to OPERABLE status given the fact that MSIV closure will result in isolation of the MSLs and a potential for plant shutdown. For



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

penetration flow paths with only one PCIV, proposed ACTION C allows 4 hours to restore an inoperable valve to OPERABLE status and 12 hours to restore EFCVs in reactor instrumentation line penetrations. The four hour Completion Time is reasonable considering the relative stability of the closed system to act as a penetration boundary and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3. The Completion Time of 12 hours is reasonable considering the instrument and the small pipe diameter of the affect penetrations. During the allowed time, a limiting event would still be assumed to be within the bounds of the safety analysis, assuming no single active failure. Allowing this extended time to potentially avoid a plant transient caused by the immediate forced shutdown is reasonable based on the probability of an event and does not represent a significant decrease in safety.

- L5 In the event both valves in a penetration are inoperable, the existing Specification, which requires maintaining one isolation valve OPERABLE, would not be met and an immediate shutdown is required. The proposed ACTION (ACTION B) provides 1 hour prior to commencing a required shutdown. This proposed 1 hour period is consistent with the proposed BWR Standard Technical Specification time allowed for conditions when the primary containment is inoperable. The proposed change will provide consistency in actions for these various containment degradations.
- L6 The frequency of the periodic verification required when a penetration has been isolated to comply with current Specification 3.7.D.2 has been changed from daily to monthly. These valves are strictly controlled and are operated in accordance with plant procedures. Daily verification that these valves are still isolated places an undue burden on plant operations and provides little if any gain in safety, since these valves are rarely found in the unisolated condition, once closed. Note that CTS 4.7.D.2 requires the position of one other valve in the line be "recorded" daily versus the ISTS wording of "verified." ISTS also allows an inoperable valve to be used for isolating the penetration.
- L7 The Note to SR 3.6.1.3.1 allows the SR to not be met (i.e., do not have to verify closed) when the valves are open for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry and for Surveillances that require the valves to be open. For these reasons, it is deemed acceptable to open the valves for short periods of time. CTS 3.7.F.3.a, which allows the 18-inch primary containment isolation valves associated with PURGING to be open during the RUN mode during a 24-hour period after entering the RUN mode and/or

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

for a 24-hour period prior to entering the SHUTDOWN mode, is encompassed by the provisions of the Note. The additional exemptions allowed by the Note are acceptable since the 18-inch purge valves continue to be capable of closing in the environment following a LOCA.

- L8 The time allowed to shutdown the plant when the required actions are not met has been changed from "in the COLD SHUTDOWN CONDITION within 24 hours" to in MODE 3 (Hot Shutdown) in 12 hours and MODE 4 (Cold Shutdown) within 36 hours. The proposed allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The additional 12 hours allowed to reach Mode 4 is offset by the safety benefit of being subcritical (MODE 3) in a shorter required time.
- L9 This change adds proposed ACTION D which relaxes the allowed outage time from 4 hours to 8 hours to isolate the affected penetration if one main steam isolation valve (MSIV) in one or more penetrations is inoperable (due to leakage or other reason). This will allow a longer period of time to restore the MSIVs to OPERABLE status in order to prevent the potential for a plant shutdown by isolating the main steam line(s). During the additional time allowed, a limiting event would still be assumed to be within the bounds of the safety analysis, assuming no single active failure. Allowing this extended time to potentially avoid a plant transient caused by a plant shutdown is reasonable and does not represent a significant decrease in safety.



UNIT 1

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

2007 / 10 / 14



~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A PRIMARY CONTAINMENT~~

~~4.7.A PRIMARY CONTAINMENT~~

~~3. Pressure Suppression Chamber
Reactor Building Vacuum Breakers~~

~~3. Pressure Suppression Chamber
Reactor Building Vacuum Breakers~~

Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.

The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.

M1

Applicability Modes 2 + 3

SR 3.6.1.5.3

A2

Proposed Note for ACTIONS

ACTIONS A & C

L1

Proposed ACTIONS B, D & E

L1

From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.

SR 3.6.1.5.3

L2

SR 3.6.1.5.3

LAI

A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

M2

Proposed SR 3.6.1.5.1

~~4. Drywell-Pressure Suppression Chamber Vacuum Breakers~~

~~4. Drywell-Pressure Suppression Chamber Vacuum Breakers~~

When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c., below.
One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.
When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

BFN Unit 1

See Justification for Changes for BFN ISTS 3.6.1.6

3.7/4.7-10

AMENDMENT NO. 2 2 2



L2

TABLE 3.7.A
INSTRUMENTATION FOR CONTAINMENT SYSTEMS

Minimum No.
Operable Per
Trip System

2

Function

Instrument Channel -
Pressure suppression
chamber-reactor building
vacuum breakers
(PdIS-64-20, 21)

Trip Level Setting

0.5 psid

Action

(1)

Remarks

Actuates the pressure
suppression chamber-
reactor building
vacuum breakers.

Footnote:

(1) - Repair in 24 hours. If the function is not OPERABLE in 24 hours, declare the system or component inoperable.

3



L2

TABLE 4.7.A

CONTAINMENT SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>Function</u>	<u>Functional Test</u>	<u>Calibration</u>	<u>Instrument Check</u>
Instrument Channel- Pressure suppression chamber reactor building vacuum breakers (Fdis-64-20, 21)	Once/month (1)	Once/18 months (2)	None

Footnotes:

- (1) - Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify OPERABILITY of the trip and alarm functions.
- (2) - Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the level setting.

REF
Unit 1

3.7/4.7-24b

AMENDMENT NO. 222
PAGE 4 OF 4

JUL 17 1995

Specification 3.6.1.5



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.A Primary Containment (AI)

4.7.A Primary Containment

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

(AI)
a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.

SR 3.6.1.5.

a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.

(MI)
Applicability Modes 1, 2 & 3

SR 3.6.1.5.3

SR 3.6.1.5.3

(A2) Proposed Note for ACTIONS →

b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.

SR 3.6.1.5.3

b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

(LI)
ACTIONS A+C

(LI)
Proposed ACTIONS B, D+E

(M2)
Proposed SR 3.6.1.5.1 →

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c., below.
b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.
b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

See Justification for Changes for BFN ISTS 3.6.1.6



42

TABLE 3.7.A

INSTRUMENTATION FOR CONTAINMENT SYSTEMS

Minimum No.
Operable Per
Trip System

2

Function

Instrument Channel -
Pressure suppression
chamber-reactor building
vacuum breakers
(PdIS-64-20, 21)

Trip Level Setting

0.5 psid

Action

(1)

Remarks

Actuates the pressure
suppression chamber-
reactor building
vacuum breakers.

Footnote:

(1) - Repair in 24 hours. If the function is not OPERABLE in 24 hours, declare the system or component inoperable.

BFN
Unit 2

3.7/4.7-24a

AMENDMENT NO. 237
PAGE 3 OF 4

Specification 3.6.1.5
JUL 17 1995



TABLE 4.7.A

CONTAINMENT SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>Function</u>	<u>Functional Test</u>	<u>Calibration</u>	<u>Instrument Check</u>
Instrument Channel- Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21)	Once/month ⁽¹⁾	Once/18 months ⁽²⁾	None

Footnotes:

- (1) - Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify OPERABILITY of the trip and alarm functions.
- (2) - Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the level setting.

L2

BRN
Date 2

3.7/4.7-24b

PAGE 4 OF 4

AMENDMENT NO. 237

Specification 3.6.15
JUL 17 1985

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~2.7.A PRIMARY CONTAINMENT~~

(A1)

~~3. Pressure Suppression Chamber
Reactor Building Vacuum Breakers~~

a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.

(M1) Applicability notes 1, 2, 3

SR 3.6.1.5.3

(A2) Proposed Note for Actions

(L1) Actions A+C

(L1) Proposed Actions B, D, +E

b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c below.
b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

~~4.7.A PRIMARY CONTAINMENT~~

~~3. Pressure Suppression Chamber
Reactor Building Vacuum Breakers~~

SR 3.6.1.5

a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.

SR 3.6.1.5.3

(L2)

SR 3.6.1.5.3

(LAI)

b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

(M2)

Proposed SR 3.6.1.5.1

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.
b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

L2

TABLE 3.7.A

INSTRUMENTATION FOR CONTAINMENT SYSTEMS

Minimum No. Operable Per Trip System	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21)	0.5 psid	(1)	. Actuates the pressure suppression chamber-reactor building vacuum breakers.

Footnote:

(1) - Repair in 24 hours. If the function is not OPERABLE in 24 hours, declare the system or component inoperable.

BFN
Under 3

3.7/4.7-23b

AMENDMENT NO. 196

3 OF 4

Specification 3.6.1.5
JUL 17 1995

TABLE 4.7.A

CONTAINMENT SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>Function</u>	<u>Functional Test</u>	<u>Calibration</u>	<u>Instrument Check</u>
Instrument Channel- Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21)	Once/month ⁽¹⁾	Once/18 months ⁽²⁾	None

Footnotes:

- (1) - Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify OPERABILITY of the trip and alarm functions.
- (2) - Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the level setting.

(12)

Specification 3.6.1.5
JUL 17 1995

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.5
REACTOR BUILDING-TO-SUPPRESSION CHAMBER VACUUM BREAKERS

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 Existing LCO 3.7.A.3 is being replaced by proposed LCO 3.6.1.6. The proposed LCO will contain a Note stating that: "Separate Condition entry is allowed for each line." This note clarifies that the Conditions and Required Actions that follow may be applied to each of the two reactor building-to-suppression chamber vent paths without regard to vent path status. Each vent path contains two vacuum breakers in series. This note provides directions consistent with the intent of the Required Actions. This change is consistent with NUREG-1433.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 CTS 3.7.A.2.a requires the primary containment to be OPERABLE at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the vessel. The proposed BFN ISTS 3.6.1.6 applicability is MODES 1, 2, and 3. This is more restrictive since CTS does not require the primary containment to be OPERABLE when in MODE 2, not critical and < 212°F.
- M2 A new Surveillance Requirement has been added to verify each vacuum breaker is closed (except when they are open for performance of Surveillances) every 14 days. This is consistent with the BWR Standard Technical Specifications, NUREG 1433.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.5
REACTOR BUILDING-TO-SUPPRESSION CHAMBER VACUUM BREAKERS

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 Details of visual inspections of valves have been relocated to plant procedures. This type of inspection is more appropriately controlled by plant procedures. The valves are still required by Technical Specifications to be cycled and their setpoint verified to ensure operability. Any changes to procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 Existing LCO 3.7.A.3.b identifies the currently required actions if one reactor building-to-suppression chamber vacuum breaker is inoperable. If more than one vacuum breaker is inoperable, the existing specification assumes either containment integrity is lost or the ability to relieve negative pressure in the containment is lost. Therefore, LCO 3.7.A.3.b. defaults to 1.0.C.1 which requires that the reactor be placed in Hot Standby within 6 hours and Cold Shutdown within the following 30 hours. Proposed LCO 3.6.1.6 recognizes that there are two vacuum breakers in series in each of two vent paths between the reactor building and suppression chamber. As a result, if one vacuum breaker in each vent path is not closed (Condition A), containment integrity and venting capability are still maintained and 7 days is provided to restore the redundancy for containment integrity in each vent line. Likewise, if two vacuum breaker valves in one vent line are inoperable but closed (Condition C), containment integrity and venting capability are still maintained and 7 days is provided to restore the redundant vent path. Therefore, proposed Specification 3.6.1.6 makes the distinction between loss of redundancy and loss of function. The existing specification fails to make this distinction between loss of function and loss of redundancy and, therefore, is unnecessarily conservative. In addition, loss of function (loss of containment integrity (Condition B) or loss of venting capability (Condition D)) will require initiating action within 1 hour instead of immediately. Also, CTS 3.7.A.3 does not have a specific shutdown requirement, therefore, CTS 1.0.C.1 applies. CTS 1.0.C.1 requires the unit be placed in Hot Standby within 6 hours and Cold Shutdown within the following 30 hours. Proposed ACTION E requires the Unit to be placed in Hot Shutdown with 12 hours and Cold Shutdown within 36 hours. Proposed ACTION E is considered less restrictive since additional time is allowed prior to requiring the plant to be in a lesser Mode (i.e., Proposed Action E requirement to be in Hot Shutdown in 12 hours versus the CTS requirement to be in Hot Standby in 6 hours). This change is consistent with NUREG-1433.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.5
REACTOR BUILDING-TO-SUPPRESSION CHAMBER VACUUM BREAKERS

- L2 The vacuum breaker actuation instrumentation Surveillances are proposed to be deleted from Technical Specifications. The requirement of SR 3.6.1.5.3 to ensure the vacuum breakers are full open at 0.5 psid is sufficient. Vacuum breaker actuation instrumentation is required to be OPERABLE to satisfy the setpoint verification Surveillance Requirement (SR 3.6.1.5.3) for the vacuum breakers. If the vacuum breaker actuation instrumentation is inoperable, then the Surveillance Requirement cannot be satisfied and the appropriate actions must be taken for inoperable vacuum breakers in accordance with the ACTIONS of Specification 3.6.1.5. As a result, the requirements for the vacuum breaker actuation instrumentation are adequately addressed by the requirements of Specification 3.6.1.5 and SR 3.6.1.5.3 and are proposed to be deleted from Technical Specifications.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



(A1)

3.7.A PRIMARY CONTAINMENT

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

- a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.
- b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.

4.7.A PRIMARY CONTAINMENT

3. Pressure Suppression Chamber- Reactor Building Vacuum Breakers

- a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.
- b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

See Justification for Changes for BFN ISTS 3.6.1.5

~~4. Drywell Pressure Suppression Chamber Vacuum Breakers~~

(A1)
 (M1)
 Applicability Modes 1, 2 & 3
 LCO 3.6.1.6
 When performing their intended function

- a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c, below. (A1)

- b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

Note 2 to SR 3.6.1.6.1

BFN Unit 1

(L1) Proposed ACTION A → 3.7/4.7-10
 (L2) Proposed ACTION B →

~~4. Drywell Pressure Suppression Chamber Vacuum Breakers~~

SR 3.6.1.6.2

- a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.

Proposed SR 3.6.1.6.1 →

- b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

AMENDMENT NO. 222



(A1)

~~3.7.4.7-11~~
~~OPERATING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

~~4.7.A Primary Containment~~

~~3.7.A.4 (Cont'd)~~

~~4.7.A.4 (Cont'd)~~
SR 3.6.1.6.3

LC03.6.1.6 c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.

c. Each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.3.4.

See Justification for Changes for BFN 1ST3 B.6.1.1

ACTION C
d. If Specifications 3.7.A.4.a, 3.7.A.4.b, or 3.7.A.4.c cannot be met, the unit shall be placed in a COLD SHUTDOWN CONDITION in an orderly manner within 24 hours.
36-44
In a HOT SHUTDOWN Condition in 12hrs and

d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

M3

5. Oxygen Concentration

5. Oxygen Concentration

- a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.
- b. Within the 24-hour period subsequent to placing the reactor in the RUN MODE following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.
- c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a COLD SHUTDOWN CONDITION within 24 hours.
- d. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN CONDITION within 24 hours.

- a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.
- b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.
- c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

See Justification for CHANGES FOR BFN 1ST3 3.6.3.1



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



JUL 17 1995

3.7/4.7 CONTAINMENT SYSTEMS

(A1)

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.A Primary Containment

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

- a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.
- b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.

4.7.A Primary Containment

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

- a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.
- b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

See Justification for Changes for BFN 15TS 3.6.1.5

(A1) Drywell-Pressure Suppression Chamber Vacuum Breakers

(M1)

Applicability MODES 6.2 + 3

LCO 3.6.1.6

When performing their intended function

(A2)

Note 2 to SR 3.6.1.6.1

a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c., below.

b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

(A1) Drywell-Pressure Suppression Chamber Vacuum Breakers

SR 3.6.1.6.2

(M2)

Proposed SR 3.6.1.6.1

a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.

(L3)

b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

BFN (L1) Proposed ACTION A →

Unit 2

(L2) Proposed ACTION B →

3.7/4.7-10

AMENDMENT NO. 237

PAGE 2 OF 3

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

~~4.7.A Primary Containment~~

~~3.7.A.4 (Cont'd)~~

~~4.7.A.4 (Cont'd)~~

SR 3.6.1.63

LC036.16 c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.

c. Each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches

in accordance with Specification 1.0.YM

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.1.1

ACTION 2

d. If Specifications 3.7.A.4.a, .b, or .c cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.

d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

L4 24 36

in a Hot Shutdown condition in 12 hrs

M3

5. Oxygen Concentration

5. Oxygen Concentration

a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100/psig, except as specified in 3.7.A.5:b.

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a Cold Shutdown condition within 24 hours.

c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

d. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.3.1

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

JUL 17 1985

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

(A1)

3.7.A PRIMARY CONTAINMENT

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

- a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.
- b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.

(A1)

~~4. Drywell-Pressure Suppression Chamber Vacuum Breakers~~

(M1)

Applicability Modes 1,2+3

LCO 3.6.1.6

When performing their intended function

- a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c below.

(A1)

(A2)

Note 2 to SR 3.6.1.6.1

- b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

BFN Unit 3 (L1) Proposed Action A → (L2) Proposed Action B →

3.7/4.7-10

4.7.A PRIMARY CONTAINMENT

3. Pressure Suppression Chamber- Reactor Building Vacuum Breakers

- a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.
- b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

See Justification for Changes for BFN ISTS 3.6.1.5

(A1)

~~4. Drywell-Pressure Suppression Chamber Vacuum Breakers~~

~~SR 3.6.1.6.2~~

- a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.

(M2)

Proposed SR 3.6.1.6.1 →

(L3)

- b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

AMENDMENT NO. 196

PAGE 2 OF 3

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

~~4.7.A Primary Containment~~

~~3.7.A.4 (Cont'd)~~

~~4.7.A.4 (Cont'd)~~

SR 3.6.1.6.3

LCO 3.6.1.6 c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.

c. Once each operating cycle, each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches

in accordance with Specification 1.0.MM.

Action C d. If Specifications 3.7.A.4.a, 3.7.A.4.b, or 3.7.A.4.c, cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.

d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

In a HOT SHUT DOWN condition in 12 hrs and

See Justification for Changes for BFN 15 TS 3.6.1.1

5. Oxygen Concentration

5. Oxygen Concentration

a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100/psig, except as specified in 3.7.A.5.b.

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a Cold Shutdown condition within 24 hours.

c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

d. If the specifications of 3.7.A.5.a through 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

See Justification For Changes 15 TS 3.6.3.1

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.6
SUPPRESSION-CHAMBER-TO-DRYWELL VACUUM BREAKERS

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The words "except when performing their intended function" have been added to preclude requiring the LCO to be met when the valves cycle automatically. Since their intent is to open when a sufficient differential pressure exists, this change is considered administrative only.
- A3 CTS 4.7.A.4.c is performed in accordance with the Inservice Testing Program on a frequency of every operating cycle. Proposed SR 3.6.1.6.3 is to be performed every 18 months. Since an operating cycle at BFN is approximately 18 months, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 CTS 3.7.A.2.a requires the primary containment to be OPERABLE at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the vessel. The proposed BFN ISTS 3.6.1.6 applicability is MODES 1, 2, and 3. This is more restrictive since CTS does not require the primary containment to be OPERABLE when in MODE 2, not critical and < 212°F.
- M2 A new Surveillance Requirement (proposed SR 3.6.1.6.1) has been added to verify the vacuum breakers are closed once every 14 days. This new SR ensures the "closed" requirement of the LCO statement is being met. This is an additional restriction on plant operation.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.6
SUPPRESSION-CHAMBER-TO-DRYWELL VACUUM BREAKERS

- M3 CTS 3.7.A.4.d requires an orderly shutdown be initiated and the reactor to be in the Cold Shutdown Condition within 24 hours when certain conditions can not be met. Proposed Action C will require the plant be in MODE 3 (Hot Shutdown Condition) in 12 hours and MODE 4 (Cold Shutdown Condition) in 36 hours. The addition of this intermediate step to the Cold Shutdown Condition is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Specific"

- L1 Proposed ACTION A allows 72 hours to restore an inoperable vacuum breaker to OPERABLE status, with one of the required vacuum breakers inoperable for opening. This is allowed since the remaining nine OPERABLE breakers are capable of providing the vacuum relief function. The 72 hours is considered acceptable due to the low probability of an event in conjunction with an additional failure in which the remaining vacuum breaker capability would not be adequate.
- L2 Proposed Action B allows a short time to close an open vacuum breaker since there is low probability of an event that would pressurize primary containment. An open vacuum breaker allows communication between the drywell and suppression chamber airspace and, as a result, there is the potential for suppression chamber overpressurization due to this bypass leakage if a LOCA were to occur. If vacuum breaker position indication is not reliable, an alternate method of verifying that the vacuum breakers are closed is to verify that a differential pressure of 0.5 psid between the suppression chamber and drywell is maintained for 1 hour without makeup. The required 2 hour Completion Time is considered adequate to perform this test.
- L3 Existing Specification 4.7.A.4.b requires that "When it is determined that a vacuum breaker is inoperable for opening at a time when operability is required, all other vacuum breakers shall be exercised immediately and every 15 days thereafter until the inoperable vacuum breaker has been returned to normal service." This requirement is not included in NUREG-1433 and will be deleted. This change eliminates the requirement to demonstrate the OPERABILITY of the redundant vacuum breakers whenever a vacuum breaker is declared inoperable. This change acknowledges that the inoperability of a vacuum breaker is not automatically indicative of a similar condition in the redundant vacuum breakers unless a generic failure is suspected and that the periodic

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.1.6
SUPPRESSION-CHAMBER-TO-DRYWELL VACUUM BREAKERS

frequencies specified to demonstrate OPERABILITY have been shown to be adequate to ensure equipment OPERABILITY. Therefore, this change allows credit to be taken for normal periodic surveillance as a demonstration of OPERABILITY and availability of the remaining components and reduces unnecessary challenges and wear to redundant components. This change is consistent with NUREG-1433.

- L4 CTS 3.7.A.4.d requires the unit to be placed in a Cold Shutdown condition in an orderly manner within 24 hours. Proposed ACTION C is less restrictive since it requires the unit to be placed in MODE 3 (Hot Shutdown condition) in 12 hours and in MODE 4 (Cold Shutdown condition) in 36 hours. The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



(A1)

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

Applicability

LCD 3.6.2.1

a. Minimum water level =
 -6.25" (differential pressure control >0 psid)
 -7.25" (0 psid differential pressure control)

b. Maximum water level =
 -1"

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

1. Pressure Suppression Chamber

a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECSS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

See Justification for Changes for BFN 1STS 3.5.2

See Justification for CHANGES for BFN 3.6.2.2

(M2)

Add SR 3.6.2.1.1 firs-1 frequency -once/24 hours



AUG 23 1991

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

A1

3.7.A. Primary Containment

Proposed Required Action A1

M3

3.7.A.1 (Cont'd)

ACTION A

Condition A

c. With the suppression pool water temperature > 95°F initiate pool cooling and restore the temperature to ≤ 95°F within 24 hours or be in at least HOT SHUTDOWN CONDITION within the next 6 hours and in the COLD SHUTDOWN CONDITION within the following 30 hours.

LCD 3.6.2.1.a

Required Action A.2

L1

ACTION B

Required Action D.3

When any OPERABLE TRM Channel is ≤ 25/40 divisions of full scale on Range 7

ACTION C

Condition C

d. With the suppression pool water temperature > 105°F during testing of ECCS or relief valves, stop all testing, initiate pool cooling and follow the action in Specification 3.7.A.1.c above.

LCD 3.6.2.1.b

Required Action C.1

LA1

ACTION A

ACTION D

Condition D

e. With the suppression pool water temperature > 110°F during the STARTUP CONDITION, HOT STANDBY CONDITION (with all control rods not inserted), or REACTOR POWER OPERATION, the reactor shall be scrammed.

LCD 3.6.2.1.c

Required Action D.1

Applicability Modes 1, 2 + 3

M4

Proposed Required Action D.2

ACTION E

Condition E

f. With the suppression pool water temperature > 120°F following reactor isolation, depressurize to < 200 psig at normal cooldown rates.

M5

M4

UNIT 2

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



~~3.7/A.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

(A1)

~~Specification~~

~~A. Primary Containment~~

- 1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

Applicability

LC0 3.6.2.1

- a. Minimum water level = -6.25" (differential pressure control >0 psid)
-7.25" (0 psid differential pressure control)

- b. Maximum water level = -1"

(A1)

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

~~Specification~~

~~A. Primary Containment~~

~~1. Pressure Suppression Chamber~~

- a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

SEE JUSTIFICATION FOR CHANGES FOR BFN 1153.5.2

SEE JUSTIFICATION FOR CHANGES FOR BFN 3.6.2.2

(M2)

Add SR 3.6.2.1.1 first frequency - once / 24 hours



LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.A. Primary Containment

3.7.A.1 (Cont'd)

Proposed Required Action A 1

M3

ACTION A
Condition A

Required Action A.2

c. With the suppression pool water temperature > 95°F initiate pool cooling and restore the temperature to ≤ 95°F within 24 hours or be in at least the HOT SHUTDOWN CONDITION within the next 6 hours and in the COLD SHUTDOWN CONDITION within the following 30 hours

LCO 3.6.2.1.a

L1

ACTION B

Required Action B.3

When any OPERABLE ILM channel is ≤ 25/40 divisions of full scale on Range 7

L1

ACTION C

Condition C

Required Action C.1

d. With the suppression pool water temperature > 105°F during testing of ECCS or relief valves, stop all testing, initiate pool cooling and follow the action in Specification 3.7.A.1.c above.

LCO 3.6.2.1.b

LAI

ACTION A

ACTION D

Condition D

Required Action D.1

e. With the suppression pool water temperature > 110°F during the STARTUP CONDITION, HOT STANDBY CONDITION (with all control rods not inserted), or REACTOR POWER OPERATION, the reactor shall be scrammed.

LCO 3.6.2.1.c

Applicability Modes 1, 2 + 3

M1

Proposed Required Action D 2

M4

ACTION E

Condition E

M5

M4

f. With the suppression pool water temperature > 120°F following reactor isolation, depressurize to < 200 psig at normal cooldown rates.



UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

(A1)

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

(A1)

~~Specification~~

~~A. Primary Containment~~

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

Applicability

LCD 3.6.2.1

a. Minimum water level =
-6.25" (differential pressure control >0 psid)
-7.25" (0 psid differential pressure control)

b. Maximum water level =
-1"

See Justification for changes for BFN 1ST 3.5.2

See Justification for changes for BFN 3.6.2.2

(M2) Add SR 3.6.2.1.1 first frequency - once / 24 hours

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

~~Specification~~

~~A. Primary Containment~~

(A1)

~~1. Pressure Suppression Chamber~~

a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.



3.7.A. Primary Containment

3.7.A.1 (Cont'd)

Proposed Required Action A1

M3

c. With the suppression pool water temperature > 95°F initiate pool cooling and restore the temperature to ≤ 95°F within 24 hours or, be in at least the HOT SHUTDOWN CONDITION within the next 6 hours and in the COLD SHUTDOWN CONDITION within the following 30 hours.

LCO 3.6.2.1.a

L1

When any OPERABLE Irm channel is ≤ 25/40 division of full scale on Range 7

Action A Condition A

Required Action A2

L1

Action B

A1

Required Action D.3

d. With the suppression pool water temperature > 105°F during testing of ECCS or relief valves, stop all testing, initiate pool cooling and follow the action in Specification 3.7.A.1.c above.

LCO 3.6.2.1.b

Action C Condition C

Required Action C.1

Action A

LAI

Action D

Condition D

Required Action D.1

e. With the suppression pool water temperature > 110°F during the STARTUP CONDITION, HOT STANDBY CONDITION (with all control rods not inserted), or REACTOR POWER OPERATION the reactor shall be scrammed.

LCO 3.6.2.1.c

Applicability Mode 1, 2+3

M1

Proposed Required Action D.2

M4

f. With the suppression pool water temperature > 120°F following reactor isolation, depressurize to < 200 psig at normal cooldown rates.

Action E

Condition E

M5

M4



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.1 - SUPPRESSION POOL AVERAGE TEMPERATURE

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 Existing Specification 3.7.A.1.e modifies the applicability governing suppression pool temperature such that the temperature limit applies only during the STARTUP CONDITION, HOT STANDBY CONDITION (with all control rods inserted), or REACTOR POWER OPERATION. Proposed LCO 3.6.2.1, Suppression Pool Average Temperature, ACTION D is applicable in Modes 1, 2, and 3. Therefore, this change is more restrictive.
- M2 CTS Surveillance Requirement 4.7.A.1.a only requires continual suppression pool temperature monitoring and logging whenever heat is added to the suppression pool during testing. Proposed SR 3.6.2.1.1 is more restrictive since it also requires this verification be performed once every 24 hours in the absence of testing.
- M3 A new Required Action has been added (proposed Required Action A.1) to verify temperature is $\leq 110^{\circ}\text{F}$ every hour, anytime temperature has exceeded 95°F . This is an additional restriction on plant operation.
- M4 When temperature exceeds 110°F , the current requirements only require the reactor to be scrammed. Proposed Required Action D.2 requires the temperature to be verified $\leq 120^{\circ}\text{F}$ every 30 minutes and a cooldown to MODE 4 within 36 hours, respectively. If temperature exceeds 120°F , the current requirements only require the RPV to be depressurized to < 200 psig at normal cooldown rates. Proposed ACTION E now requires the 200 psig limit to be attained in 12 hours, and to continue cooling down the plant to cold shutdown (MODE 4) within 36 hours. These are additional restrictions on plant operation.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.1 - SUPPRESSION POOL AVERAGE TEMPERATURE

- M5 The proposed ACTION (ACTION E) when pool temperature exceeds 120°F does not depend upon whether the reactor is isolated. If pool temperature reaches 120°F, regardless of whether the reactor is isolated, significant heat could still be added to the suppression pool and the Required Action is appropriate. Even with the reactor not isolated, there may be no heat rejection from the containment, as in the case of loss of condenser vacuum. Applying the actions regardless of whether the reactor is isolated does not introduce any operation which is unanalyzed. This change is more restrictive on plant operations.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 Details of how to reduce suppression pool temperature to within the limits have been relocated to plant procedures. Methods for restoring pool temperature are more appropriately located in plant procedures. Changes to the procedure will be controlled by the licensee controlled programs.

"Specific"

- L1 The Applicability for proposed LCO 3.6.2.1, Suppression Pool Average Temperature, is Modes 1, 2, and 3. However, this Applicability is modified within LCO 3.6.2.1 so that a lower suppression pool temperature limit applies if any Operable IRM channel is on Range 7 or above. This limit was selected so that the suppression pool temperature limits are applicable when the reactor is critical with reactor power approximately at the point of adding heat. As a result of this qualification to the Applicability statement, suppression pool temperature is required to be maintained at a temperature of less than 95°F (or less than 105°F while performing tests that add heat to the suppression pool) only when the reactor is critical with reactor power at the approximate level where heat generated is approximately equal to normal system heat losses. If the reactor is not critical or at a power below the point of adding heat, the suppression pool may be maintained at an average temperature up to 110°F. This change is less restrictive because CTS 3.7.A.1. required the lower suppression pool temperature to be less than 95°F (or less than 105°F while performing tests that add heat to the suppression pool) even if the reactor is not critical or not above the point of adding heat. If the reactor is not critical or the reactor is below the point of adding heat, there is significantly less heat generation from decay heat than assumed in the design basis. The suppression pool is designed to absorb the decay heat and sensible energy released during a reactor blowdown via safety/relief valves or from design basis accidents when the reactor has been operating continuously at full power for a



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.1 - SUPPRESSION POOL AVERAGE TEMPERATURE

considerable period of time. Any event initiated with reactor power or reactor power history less than these conditions will place considerably less heat load on the suppression pool than a DBA LOCA. This change is consistent with NUREG-1433. In addition, the shutdown requirements, if the temperature is not restored, have been modified to only require reducing power to below IRM Range 7 within 12 hours, consistent with the new Applicability.

UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

(A1)

~~Specification~~

A. Primary Containment

1. Applicability At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

LCO 3.6.2.2

- a. Minimum water level =
 - 6.25" (differential pressure control >0 psid)
 - 7.25" (0 psid differential pressure control)
- b. Maximum water level = -1"

(L1) Proposed ACTION A

(L2) Proposed ACTION B

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

~~Specification~~

~~A. Primary Containment~~

1. Pressure Suppression Chamber
SR 3.6.2.2.1

a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

[See Justification for Changes for BFN ISTS 3.5.2]

[See Justification for Changes for BFN ISTS 3.6.2.1]



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



See Justification for Changes for BFN ISTS 3.6.2.1

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

Applicability

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure

or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

- a. Minimum water level = -6.25" (differential pressure control >0 psid)
-7.25" (0 psid differential pressure control)
- b. Maximum water level = -1"

(L1) Proposed ACTION A

(L2) Proposed ACTION B

LC 3.6.2.2

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

(A1)

1. Pressure Suppression Chamber

SR 3.6.2.2.1

- a. The suppression chamber water level be checked once per day.

Whenever heat is added to the suppression pool by testing of the ECOS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.5.2

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.2.1

UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

3.7/4.7 CONTAINMENT SYSTEMS

See Justification for changes for BFN ISTS 3.6.2.1

Specification 3.6.2.2

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

Applicability 1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

LCO 3.6.2.2

- a. Minimum water level = -6.25" (differential pressure control >0 psid)
-7.25" (0 psid differential pressure control)
- b. Maximum water level = -1"

- (L1) Proposed Action A
- (L2) Proposed Action B

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

1. Pressure Suppression Chamber

SR 3.6.2.2.1

- a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

See Justification for changes for BFN ISTS 3.5.2

See Justification for changes for BFN ISTS 3.6.2.1

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.2 - SUPPRESSION POOL WATER LEVEL

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Specific"

L1 The existing Action for suppression pool water level outside limits (Specification 3.7.A.1) allows no time to restore level. An unanticipated change in suppression pool level would require addressing the cause and aligning the appropriate system to raise or lower the pool level. These activities require some time to accomplish without undo haste. The out-of-service time is based on engineering judgement of the relative risks associated with: 1) the safety significance of the system; 2) the probability of an event requiring the safety function of the system; and 3) the relative risks associated with the plant transient and potential challenge of safety systems experienced by requiring a plant shutdown. Upon further review, and discussion with the NRC Staff, during the development of the BWR Standard Technical Specifications, NUREG 1433, a 2 hour restoration allowance was determined to be appropriate.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.2 - SUPPRESSION POOL WATER LEVEL

- L2 Per CTS, if suppression pool water level is not maintained within limits, the Specification is violated and in accordance with TS 1.0.C.1 the plant must be placed in Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours unless suppression pool water level is restored. This provides actions for circumstances not directly provided for in the specifications and where occurrence would violate the intent of the specification. The BFN ISTS provides Action within the Specification which could be considered less restrictive than CTS. Action B allows 12 hours to be in MODE 3 (Hot Shutdown) and 36 hours to be in MODE 4 (Cold Shutdown). The proposed Action is considered less restrictive since 12 hours is allowed to place the unit in Hot Shutdown versus the 6 hours allowed to place the unit in Hot Standby per CTS.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



(A1)

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~
LCO 3.6.2.3

1. The RHRS shall be OPERABLE #:

Applicability

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

See Justification Changes for BFN 15TS 3.5.1

~~4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Test Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



AUG 02 1989

~~LIMITING CONDITION FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

ACTION A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

(A1) ~~5. No additional surveillance required.~~

See Justification for Changes for BFN 1STS 3.8.1

ACTION B

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

(A1) ~~6. No additional surveillance required.~~

Add SR 3.6.2.3.1
SR 3.6.2.3.2

MI

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

(A1) ~~7. No additional surveillance required.~~

See Justification For Changes For BFN 1STS 3.6.2.4 + 3.6.2.5

Proposed ACTION C

(L)



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

3.5.B ~~Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

4.5.B ~~Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

~~8. No additional surveillance required.~~

ACTION
D

HOT SHUTDOWN
CONDITIONS
in 12 hours
and

M2

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within ²⁴~~36~~ hours. (L2)

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. No additional surveillance required.

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification For Changes for BFN 15 TS 3.5.1 + 3.5.2

UNIT 2

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

LCO 3.6.2.3

1. The RHRS shall be OPERABLE #:

Applicability

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

See Justification for Changes for BFN ISTS 3.5.1

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Testable Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

~~3.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

ACTION A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

(A1)

~~5. No additional surveillance required.~~

See Justification For Changes for BFN ISTS 3.8.1

ACTION B

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

(A1)

~~6. No additional surveillance required.~~

Add SR 3.6.2.3.1
SR 3.6.2.3.2

(M1)

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

(A1)

~~7. No additional surveillance required.~~

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.2.4 AND 3.6.2.5

(L1)

Proposed ACTION C



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

3.5.B ~~Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

4.5.B ~~Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

~~8. No additional surveillance required.~~

ACTION D

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within ~~24~~ ³⁶ hours.

(M2)

HOT SHUTDOWN CONDITION in 12 hrs. and

(L2)

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.5.1 + 3.5.2

UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

LCO 3.6.2.3

1. The RHRS shall be OPERABLE #:

Applicability

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

See Justification for changes for BFN ISTS 3.5.1

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle.
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Testable Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

~~LIMITING CONDITION FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

Action A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

(A1)

~~5. No additional surveillance required.~~

See Justification for Changes for BFN 1STS 3.8.1

Action B

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

(A1)

~~6. No additional surveillance required.~~

Add SR 3.6.2.3.1
SR 3.6.2.3.2 (m)

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

(A1)

~~7. No additional surveillance required.~~

See Justification for Changes for BFN 1STS 3.6.2.4 And 3.6.2.5

(L1)

Proposed Action C

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.5.B ~~Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~8. No additional surveillance required.~~

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours. (L2)

Action D

(M2)

HOT SHUTDOWN CONDITION IN 12 hrs

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The B and D RHR pumps on unit 2 which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

see justification for changes for BFN ISTS 3.5.1 + 3.5.2

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.3 - RHR SUPPRESSION POOL COOLING

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 Surveillance Requirements (SR 3.6.2.3.1 and 3.6.2.3.2) have been added to ensure that the correct valve lineup for the RHR suppression pool cooling subsystems is maintained and RHR pump testing is performed to ensure the RHR suppression pool cooling subsystems remain capable of providing the overall DBA suppression pool cooling requirement. This change is consistent with NUREG-1433.
- M2 CTS 3.5.B.8 requires an orderly shutdown be initiated and the reactor to be in the Cold Shutdown Condition within 24 hours when required RHR suppression pool cooling subsystems are inoperable. Proposed Action D will require the plant be in MODE 3 (Hot Shutdown Condition) in 12 hours and MODE 4 (Cold Shutdown Condition) in 36 hours. The addition of this intermediate step to the Cold Shutdown Condition is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.3 - RHR SUPPRESSION POOL COOLING

TECHNICAL CHANGES - LESS RESTRICTIVE

- L1 Proposed ACTION C will allow 8 hours to restore required RHR suppression pool cooling subsystems to operable status prior to initiating a shutdown. The proposed 8 hour Completion Time provides some time to restore the required subsystems to Operable status, yet is short enough that operating an additional 8 hours is not risk significant. Only 8 hours is allowed since there is a substantial loss of the primary containment bypass leakage mitigation function. The 8 hour restoration time is considered acceptable due to the low probability of a DBA and because alternative methods to remove decay heat from the primary containment are still available. In addition, if the required subsystem(s) are restored to Operable status prior to the expiration of the 8 hours, a unit shutdown is averted. Thus, the potential of a unit scram occurring while shutting the unit down, which then could result in a need for a subsystem when it is inoperable, has been decreased.
- L2 The time to reach MODE 4, Cold Shutdown, has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added. These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

LCo 3.6.2.4

1. The RHRS shall be OPERABLE #:

Applicability

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

See Justification Changes for BFN ISTS 3.5.1

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Test Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



AUG 02 1989

LIMITING CONDITIONS FOR OPERATION

(A)

SURVEILLANCE REQUIREMENT

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

- 2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps—containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.
- 3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.
- 4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

4.5.B.1 (cont'd)

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 gpm against an indicated system pressure of 250 psig.

- 2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

(LAI)

See Justification for Changes for BFN ISTS 3.6.2.5

- 3. No additional surveillance required.

- 4. No additional surveillance required.

See Justification for Changes for BFN ISTS 3.5.1

AUG 02 1989

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITION FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

ACTION A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

ACTION B

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

(L1) Proposed ACTION C →

~~4.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~5. No additional surveillance required.~~

See Justification for Changes for BFN 1STS 3.8.1

(A1)

~~6. No additional surveillance required.~~

Add SR 3.6.2.4.1 (M1)

(A1)

~~7. No additional surveillance required.~~

See Justification for Changes for BFN 1STS 3.6.2.3 + 3.6.2.5



(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

ACTION D.

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within ~~24~~ hours.

36 L2

(A1) ~~8. No additional surveillance required.~~

M2

In the HOT SHUTDOWN CONDITION in 12 hours and

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification for Changes for BFN ISTS 3.5.1 + 3.5.2

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

3.5.B ~~Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(AI)

LCO 3.6.2.4

1. The RHRS shall be OPERABLE #:

applicability

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

See Justification for Change for BFN ISTS 3.5.1

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

4.5.B ~~Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Testable Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



AUG 02 1989

~~LIMITING CONDITIONS FOR OPERATION~~

(AI)

~~SURVEILLANCE REQUIREMENT~~

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

- 2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.
- 3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.
- 4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

4.5.B.1 (cont'd)

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 gpm against an indicated system pressure of 250 psig.

2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

SR 36.2.4.2

(LAI)

See Justification for Changes for BFN ISTS 3.6.2.5

3. No additional surveillance required.

4. No additional surveillance required.

See Justification for Changes for BFN ISTS 3.5.1

~~LIMITING CONDITION FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ AUG 02 1989

~~3.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

ACTION A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

~~AI 5. No additional surveillance required.~~

See Justification for Changes for BFN ISTS 3.8.1

ACTION B

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

~~AI 6. No additional surveillance required.~~

Add. SR 3.6.2.4.1 MI

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

~~AI 7. No additional surveillance required.~~

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.2.3 + 3.6.2.5

LI Proposed ACTION C

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

ACTION D

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(A1) ~~No additional surveillance required.~~

in the HOT SHUTDOWN CONDITION in 12 hours and

(M2)

(L2)

(36)

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

(See Justification for Changes for BFN ISTS 3.5.1 + 3.5.2)



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

LCO 3.6.2.4

1. The RHRS shall be OPERABLE #:

Applicability

- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Testable Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

See Justification for BFN 15TS 3.5.1

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.

(A1)

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

- 2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.
- 3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.
- 4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

~~4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

4.5.B.1 (cont'd)

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 gpm against an indicated system pressure of 250 psig.

- 2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

SR 3.6.2.4.2

(LAI)

See Justification for changes for BFN ISTS 3.6.2.5

- 3. No additional surveillance required.
- 4. No additional surveillance required.

See Justification for changes for BFN ISTS 3.5.1



LIMITING CONDITION FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.5 B. ~~Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

Action A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

Action B

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

(2) Proposed Action C

4.5 B. ~~Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

5. ~~No additional surveillance required.~~ (A1)

See Justification for Changes for BFN ISTS 3.8.1

(A1) ~~6. No additional surveillance required.~~

Add SR 3.6.2.4.1 (m1)

(A1) ~~7. No additional surveillance required.~~

See Justification for Changes for BFN ISTS 3.6.2.3 & 3.6.2.5



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.5.B ~~Residual Heat Removal System (RHRG) (LPCI and Containment Cooling)~~

4.5.B ~~Residual Heat Removal System (RHRG) (LPCI and Containment Cooling)~~

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within ~~24~~ hours.

~~8. No additional surveillance required.~~

Action D

In the HOT SHUTDOWN CONDITION in 12 hours AND

L2

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The B and D RHR pumps on unit 2 which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification for Changes for BFN ISTS 3.5.1 & 3.5.2

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.4 - RHR SUPPRESSION POOL SPRAY

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 Surveillance Requirement (SR 3.6.2.4.1) has been added to ensure that the correct valve lineup for the RHR suppression pool spray subsystems is maintained. This ensures that the RHR suppression pool spray subsystems remain capable of providing the overall DBA suppression pool spray requirement. This change is consistent with NUREG-1433.
- M2 CTS 3.5.B.8 requires an orderly shutdown be initiated and the reactor to be in the Cold Shutdown Condition within 24 hours when required RHR suppression pool spray subsystems are inoperable. Proposed Action D will require the plant be in MODE 3 (Hot Shutdown Condition) in 12 hours and MODE 4 (Cold Shutdown Condition) in 36 hours. The addition of this intermediate step to the Cold Shutdown Condition is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.4 - RHR SUPPRESSION POOL SPRAY

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 Details of the methods of performing surveillance test requirements have been relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 Proposed ACTION C will allow 8 hours to restore required RHR suppression pool spray subsystems to operable status prior to initiating a shutdown. The proposed 8 hour Completion Time provides some time to restore the required subsystems to Operable status, yet is short enough that operating an additional 8 hours is not risk significant. Only 8 hours is allowed since there is a substantial loss of the primary containment bypass leakage mitigation function. The 8 hour restoration time is considered acceptable due to the low probability of a DBA and because alternative methods to remove decay heat from the primary containment are still available. In addition, if the required subsystem(s) are restored to Operable status prior to the expiration of the 8 hours, a unit shutdown is averted. Thus, the potential of a unit scram occurring while shutting the unit down, which then could result in a need for a subsystem when it is inoperable, has been decreased.
- L2 The time to reach MODE 4, Cold Shutdown has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added. These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.



UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

Applicability

- 1. The RHRS shall be OPERABLE #:
- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Test Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

See Justification for Changes for BFN ISTS 3.5.1

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



AUG 02 1989

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENT~~

3.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

- 2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps—containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.
- 3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.
- 4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

4.5.B.1 (cont'd)

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 gpm against an indicated system pressure of 250 psig.

(LAI)

SR 3.6.2.5.2

- 2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

See Justification for changes for BFN ISTS 3.6.2.4

- 3. No additional surveillance required.

- 4. No additional surveillance required.

See Justification for changes for BFN ISTS 3.5.1



AUG 02 1989

~~LIMITING CONDITION FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

ACTION A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

Proposed ACTION C

(L1)

~~4.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1) ~~5. No additional surveillance required.~~

See Justification for Changes for BFN ISTS 3.8.1

(A1) ~~6. No additional surveillance required.~~

(M) Add SR 3.6.2.5.1

(A1) ~~7. No additional surveillance required.~~

See Justification for Changes for BFN ISTS 3.6.2.3 and 3.6.2.4



~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1)

~~4.5.B Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours. (36) (L2)

(A1) 8. No additional surveillance required. (M2)

in the HOT SHUTDOWN CONDITION in 12hrs and

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification for Change for BFN ISTS 3.5.1 & 3.5.2



UNIT 2
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A)

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

Applicability

- 1. The RHRS shall be OPERABLE #:
- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

See Justification for Changes for BFN ISTS 3.5.1

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Testable Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



AUG 02 1989

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENT

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

- 2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps—containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.

- 3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.

- 4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

4.5.B.1 (cont'd)

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 gpm against an indicated system pressure of 250 psig.

- 2. An ~~air test~~ on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

SR 3.6.2.5.2

(LAI)

See Justification for Changes for BFN ISTS 3.6.2.4

- 3. No additional surveillance required.

- 4. No additional surveillance required.

See Justification for Changes for BFN ISTS 3.5.1

LIMITING CONDITION FOR OPERATION

~~SURVEILLANCE REQUIREMENTS~~

AUG 02 1989

~~3.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5 B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHRS (containment cooling mode) are OPERABLE.

~~(A1) 5. No additional surveillance required.~~

ACTION A

See Justification for Changes for BFN 15TS 3.8.1

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHRS (containment cooling mode) are OPERABLE.

~~(A1) 6. No additional surveillance required.~~

(M1) Add SR 3.6.2.5.1

ACTION B

7. If two access paths of the RHRS (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

~~(A1) 7. No additional surveillance required.~~

See Justification for Changes for BFN 15TS 3.6.2.3 and 3.6.2.4

(L1) Proposed ACTION C

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~ APR 19 1994

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

(A1)

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(A1) 8. No additional surveillance required.

in the HOT SHUTDOWN CONDITION in 12 hours and (M2)

(L2)

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The RHR pumps on the adjacent units which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.5.1 + 3.5.2



UNIT 3
CURRENT
TECHNICAL
SPECIFICATION
MARKUP

Unit 3 5



(A1)

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

Applicability

- 1. The RHRS shall be OPERABLE #:
- (1) PRIOR TO STARTUP from a COLD CONDITION; or
- (2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in Specifications 3.5.B.2, through 3.5.B.7.

- 1. a. Simulated Automatic Actuation Test Once/Operating Cycle
- b. Pump OPERABILITY Per Specification 1.0.MM
- c. Motor Operated valve OPERABILITY Per Specification 1.0.MM
- d. Pump Flow Rate Once/3 months
- e. Testable Check Valve Per Specification 1.0.MM
- f. Verify that each valve (manual, power-operated, or automatic) in the injection flow-path that is not locked, sealed, or otherwise secured in position, is in its correct* position. Once/Month
- g. Verify LPCI subsystem cross-tie valve is closed and power removed from valve operator. Once/Month

See Justification for Changes for BFN ISTS 3.5.1

Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling with reactor steam dome pressure less than 105 psig in HOT SHUTDOWN, if capable of being manually realigned and not otherwise inoperable.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in a position for another mode of operation.



(A1)

3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

- 2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps—containment cooling mode and associated heat exchangers must remain OPERABLE) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are OPERABLE.
- 3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain OPERABLE.
- 4. If any 2 RHR pumps (LPCI mode) become inoperable, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.5.B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

4.5.B.1 (cont'd)

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 gpm against an indicated system pressure of 250 psig.

(A1)

SR
3.6.2.5.2

- 2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

See Justification for Changes for BFN ISTS 3.6.2.4

3. No additional surveillance required.

4. No additional surveillance required.

See Justification for Changes for BFN ISTS 3.5.1



~~LIMITING CONDITION FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5 B. Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

Action A

5. If one RHR pump (containment cooling mode) or associated heat exchanger is inoperable, the reactor may remain in operation for a period not to exceed 30 days provided the remaining RHR pumps (containment cooling mode) and associated heat exchangers and diesel generators and all access paths of the RHR (containment cooling mode) are OPERABLE.

Action B

6. If two RHR pumps (containment cooling mode) or associated heat exchangers are inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (containment cooling mode), the associated heat exchangers, diesel generators, and all access paths of the RHR (containment cooling mode) are OPERABLE.

7. If two access paths of the RHR (containment cooling mode) for each phase of the mode (drywell sprays, suppression chamber sprays, and suppression pool cooling) are not OPERABLE, the unit may remain in operation for a period not to exceed 7 days provided at least one path for each phase of the mode remains OPERABLE.

(L1) Proposed Action C →

~~4.5 B. Residual Heat Removal System (RHR) (LPCI and Containment Cooling)~~

(A1) ~~5. No additional surveillance required.~~

See Justification for Changes for BFN ISTS 3.8.1

(A1) ~~6. No additional surveillance required.~~

(M1) Add SR 3.6.2.5.1 →

(A1) ~~7. No additional surveillance required.~~

See Justification for Changes for BFN ISTS 3.6.2.3 and 3.6.2.4



~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

~~4.5.B Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)~~

8. If Specifications 3.5.B.1 through 3.5.B.7 are not met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

(A1) ~~8. No additional surveillance required.~~

L2

36

in the HOT SHUTDOWN Condition in 12 hours and

M2

9. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be OPERABLE. The pumps' associated diesel generators must also be OPERABLE. Low pressure coolant injection (LPCI) may be considered OPERABLE during alignment and operation for shutdown cooling, if capable of being manually realigned and not otherwise inoperable.

9. When the reactor vessel pressure is atmospheric, the RHR pumps and valves that are required to be OPERABLE shall be demonstrated to be OPERABLE per Specification 1.0.MM.

10. If the conditions of Specification 3.5.A.5 are met, LPCI and containment cooling are not required.

10. No additional surveillance required.

11. When there is irradiated fuel in the reactor and the reactor is not in the COLD SHUTDOWN CONDITION, 2 RHR pumps and associated heat exchangers and valves on an adjacent unit must be OPERABLE and capable of supplying cross-connect capability except as specified in Specification 3.5.B.12 below. (Note: Because cross-connect capability is not a short-term requirement, a component is not considered inoperable if cross-connect capability can be restored to service within 5 hours.)

11. The B and D RHR pumps on unit 2 which supply cross-connect capability shall be demonstrated to be OPERABLE per Specification 1.0.MM when the cross-connect capability is required.

See Justification for Changes for BFN ISTS 3.5.1 + 3.5.2



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.5 - RHR DRYWELL SPRAY**

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 Surveillance Requirement (SR 3.6.2.5.1) has been added to ensure that the correct valve lineup for the RHR drywell spray subsystems is maintained. This ensures that the RHR drywell spray subsystems remain capable of providing the overall DBA drywell spray requirement. This change is consistent with NUREG-1433.
- M2 CTS 3.5.B.8 requires an orderly shutdown be initiated and the reactor to be in the Cold Shutdown Condition within 24 hours when required RHR drywell spray subsystems are inoperable. Proposed Action D will require the plant be in MODE 3 (Hot Shutdown Condition) in 12 hours and MODE 4 (Cold Shutdown Condition) in 36 hours. The addition of this intermediate step to the Cold Shutdown Condition is considered more restrictive since CTS does not require any action to have taken place within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.5 - RHR DRYWELL SPRAY

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 Details of the methods of performing surveillance test requirements have been relocated to the Bases and procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in proposed BFN ISTS Section 5.0 and changes to the procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 Proposed ACTION C will allow 8 hours to restore required RHR drywell cooling subsystems to operable status prior to initiating a shutdown. The proposed 8 hour Completion Time provides some time to restore the required subsystems to Operable status, yet is short enough that operating an additional 8 hours is not risk significant. Only 8 hours is allowed since their loss substantially reduces the ability to maintain primary containment within design limits. The 8 hour restoration time is considered acceptable due to the low probability of a DBA and because alternative methods to remove decay heat from the primary containment are still available. In addition, if the required subsystem(s) are restored to Operable status prior to the expiration of the 8 hours, a unit shutdown is averted. Thus, the potential of a unit scram occurring while shutting the unit down, which then could result in a need for a subsystem when it is inoperable, has been decreased.
- L2 The time to reach MODE 4, Cold Shutdown has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added. These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.



UNIT 1

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



A1

~~LIMITING CONDITIONS FOR OPERATION~~

~~CONTAINMENT REQUIREMENTS~~

~~3.7.A. Primary Containment~~

~~6. Drywell-Suppression Chamber Differential Pressure~~

LCD
3.6.2.6

a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.1 psid except as specified in (1) and (2) below:

Applicability

(1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.1 psid 24 hours prior to a scheduled shutdown.

L1

LCD
Note

(2) This differential may be decreased to less than 1.1 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system and the drywell-pressure suppression chamber vacuum breakers.

ACTIONS
A+B

b. If the differential pressure of Specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six-hour period, an orderly shutdown shall be initiated and the reactor shall be in the ~~GOLD SHUTDOWN CONDITION~~ within 24 hours.

8 hour

L2

Reduce THERMAL POWER to $\leq 15\%$ in 12 hrs



UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.A. Primary Containment

(A1)

6. Drywell Suppression Chamber Differential Pressure

LCO
3.6.2.6

a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.1 psid except as specified in (1) and (2) below:

Applicability

(1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.1 psid 24 hours prior to a scheduled shutdown.

(L1)

LCO
Note

(2) This differential may be decreased to less than 1.1 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system and the drywell-pressure suppression chamber vacuum breakers.

ACTIONS
A+B

b. If the differential pressure of Specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six hour period, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours. Reduce THERMAL POWER to $\leq 15\%$ in 12 hrs.

8 hour

(L2)

4.7.A. Primary Containment

6. Drywell Suppression Chamber Differential Pressure

SR 3.6.2.6.1

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

(A2)

12 hours



UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

~~3.7.A. Primary Containment~~

~~4.7.A. Primary Containment~~

~~6. Drywell Suppression Chamber Differential Pressure~~

~~6. Drywell Suppression Chamber Differential Pressure~~

SR 3.6.2.6.1

LCO
3.6.2.6

a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.1 psid except as specified in (1) and (2) below:

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

(A2) 12 hours

Applicability

(1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.1 psid 24 hours prior to a scheduled shutdown.

(L1)

LCO Note

(2) This differential may be decreased to less than 1.1 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system and the drywell-pressure suppression chamber vacuum breakers.

Action A+B

b. If the differential pressure of Specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six-hour period, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours. Reduce thermal power to $\leq 15\%$ in 12 hours.

8 hours

(L2)

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.2.6
DRYWELL-TO-SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

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- A2 The Frequency for verifying the pressure differential between the drywell and the suppression chamber has been changed to 12 hours from shiftly. CTS Table 1.1 defines shiftly as at least once per 12 hours. As such, this is a change in presentation only and is therefore administrative.

TECHNICAL CHANGES - LESS RESTRICTIVE

- L1 The proposed change revises the required initiation point for establishing differential pressure between the drywell and suppression chamber. By increasing the initiation point following startup to 15% rated thermal power (RTP) (CTS initiation point is operating temperature and pressure, which is about 1% RTP), the drywell pressure and temperature will have sufficient time to stabilize prior to establishing the required differential pressure. As long as reactor power is below 15% RTP, the probability of an event that generates excessive loads on primary containment occurring within the first 24 hours of a startup or within the last 24 hours before shutdown is low. 24 hours is considered a reasonable amount of time to allow plant personnel to establish the required differential pressure.
- L2 CTS 3.7.A.6.b allows 6 hours to restore the differential pressure before initiating an orderly shutdown, which requires the plant to be in Cold Shutdown within 24 hours. The proposed actions allow 8 hours to restore differential pressure and 12 hours to reduce thermal power to \leq 15% RTP. Below this power level, per the proposed Specification, the LCO is no longer applicable (See Comment L1 above).

PAGE 1 OF 1

UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



APR 29 1991

~~3.7/4.7 CONTAINMENT SYSTEMS~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.7.F. Primary Containment Purge System

4.7.F. Primary Containment Purge System

3.7.F.3 (Continued)

- . these primary containment isolation valves is governed by Technical Specification 3.7.D.
- b. Pressure control of the containment is normally performed by VENTING through 2-inch primary containment isolation valves which route effluent to the Standby Gas Treatment System. The OPERABILITY of these primary containment isolation valves is governed by Technical Specification 3.7.D.

See Justification for Changes for CTS 3.7.F/4.7.F

~~3.7.G. Containment Atmosphere Dilution System (CAD)~~

(A1)

~~4.7.G. Containment Atmosphere Dilution System (CAD)~~

LCO 3.6.3.1

1. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE with:
 - a. Two independent systems capable of supplying nitrogen to the drywell and torus.

~~1. System Operability~~

(LA1)

- a. Cycle each solenoid operated air/nitrogen valve through at least one complete cycle of full travel in accordance with Specification 1.0.MM, and at least once per month verify that each manual valve in the flow path is open.

(M1)

SR 3.6.3.1.2

SR 3.6.3.1.1

- b. A minimum supply of 2,500 gallons of liquid nitrogen per system.

SR 3.6.3.1.1

- b. Verify that the CAD System contains a minimum supply of 2,500 gallons of liquid nitrogen twice per week.

(L1)

DEC 07 1994

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

A1

~~3.7.G. Containment Atmosphere Dilution System (CAD)~~

~~4.7.G. Containment Atmosphere Dilution System (CAD)~~

LCD 3.6.3.1 + Applicability

2. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE whenever the reactor is in the RUN MODE. or Startup Mode M2

2. When FCV 84-8B is inoperable, each solenoid operated air/nitrogen valve of System B shall be cycled through at least one complete cycle of full travel and each manual valve in the flow path of System B shall be verified open at least once per week.

A3

ACTIONS A

3. If one system is inoperable, the reactor may remain in operation for a period of 30 days provided all active components in the other system are OPERABLE.

Note to Required Action A.1 A2

ACTION B

4. If Specifications 3.7.G.1 and 3.7.G.2, or 3.7.G.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the ~~COLD SHUTDOWN CONDITION~~ within 24 hours.

MODE 3 within 12 hours M3

5. Primary containment pressure shall be limited to a maximum of 30 psig during repressurization following a loss of coolant accident.

LA 2

A3

6. System A may be considered OPERABLE with FCV 84-8B inoperable provided that all active components in System B and all other active components in System A are OPERABLE.
7. Specifications 3.7.G.6 and 4.7.G.2 are in effect until the first Cold Shutdown of unit 1 after July 20, 1984 or until January 17, 1985 whichever occurs first.



UNIT 2

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.7.F. Primary Containment Purge System

3.7.F.3 (Continued)

these primary containment isolation valves is governed by Technical Specification 3.7.D.

- b. Pressure control of the containment is normally performed by VENTING through 2-inch primary containment isolation valves which route effluent to the Standby Gas Treatment System. The OPERABILITY of these primary containment isolation valves is governed by Technical Specification 3.7.D.

4.7.F. Primary Containment Purge System

See Justification for Changes for CTS 3.7.F/4.7.F

3.7.G. Containment Atmosphere Dilution System (CAD)

(A1)

LCO 3.6.3.1 1. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE with:

- a. Two independent systems capable of supplying nitrogen to the drywell and torus.

- SR 3.6.3.1.1 b. A minimum supply of 2,500 gallons of liquid nitrogen per system.

4.7.G. Containment Atmosphere Dilution System (CAD)

1. System Operability

(LAI)

- a. Cycle each solenoid operated air/nitrogen valve through at least one complete cycle of full travel in accordance with Specification 1.0.MM.

(M1)

SR 3.6.3.1.2

and at least once per month verify that each manual valve in the flow path is open.

SR 3.6.3.1.1

- b. Verify that the CAD System contains a minimum supply of 2,500 gallons of liquid nitrogen twice

(L1)

per week.



DEC 07 1994

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.G. Containment Atmosphere Dilution System (CAD)~~

~~4.7.G. Containment Atmosphere Dilution System (CAD)~~

LCO 3.6.3.1
Applicability

2. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE whenever the reactor is in the RUN MODE or STARTUP MODE

(A1)

ACTION A

3. If one system is inoperable, the reactor may remain in operation for a period of 30 days provided all active components in the other system are OPERABLE.

Note to Required Action A.1 (A2)

ACTION B

4. If Specifications 3.7.G.1 and 3.7.G.2, or 3.7.G.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

Mode 3 within 12 hours (M3)

5. Primary containment pressure shall be limited to a maximum of 30 psig during repressurization following a loss of coolant accident.

LA2



UNIT 3

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP

~~LIMITING CONDITIONS FOR OPERATION~~ (A1)

~~SURVEILLANCE REQUIREMENTS~~

3.7.F. Primary Containment Purge System

4.7.F. Primary Containment Purge System

3.7.F.3 (Continued)

these primary containment isolation valves is governed by Technical Specification 3.7.D.

- b. Pressure control of the containment is normally performed by VENTING through 2-inch primary containment isolation valves which route effluent to the Standby Gas Treatment System. The OPERABILITY of these primary containment isolation valves is governed by Technical Specification 3.7.D.

See Justification for Changes for CTS 3.7.F/4.7.F

~~3.7.G. Containment Atmosphere Dilution System (CAD)~~ (A1)

~~4.7.G. Containment Atmosphere Dilution System (CAD)~~

LLD 3.6.3.1

1. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE with:

- a. Two independent systems capable of supplying nitrogen to the drywell and torus.

SR 3.6.3.1.1

- b. A minimum supply of 2,500 gallons of liquid nitrogen per system.

~~1. System Operability~~

(LA1)

- a. Cycle each solenoid operated air/nitrogen valve through at least one complete cycle of full travel in accordance with Specification 1.0.MM, and at least once per month verify that each manual valve in the flow path is open.

(M1)

SR 3.6.3.1.2

SR 3.6.3.1.1

- b. Verify that the CAD System contains a minimum supply of 2,500 gallons of liquid nitrogen twice per week.

(L1)



~~3.7.G. Containment Atmosphere Dilution System (CAD)~~

(A1)

~~4.7.G. Containment Atmosphere Dilution System (CAD)~~

2. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE whenever the reactor is in the RUN MODE, A or STARTUP MODE (M2)

LCO 3.6.3.1
+
Applicability

(A2)

Note to Required Action A.1

3. If one system is inoperable, the reactor may remain in operation for a period of 30 days provided all active components in the other system are OPERABLE.

Action A

4. If Specifications 3.7.G.1 and 3.7.G.2, or 3.7.G.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the ~~COLD SHUTDOWN CONDITION~~ within 24 hours.

Action B

(M3)

MODE 3
within 12 hours

5. Primary containment pressure shall be limited to a maximum of 30 psig during repressurization following a loss of coolant accident.

(LA2)

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.3.1 - CONTAINMENT AIR DILUTION SYSTEM

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

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A2 A NOTE was added specifying LCO 3.0.4 is not applicable. Since the current Technical Specifications do not have LCO 3.0.4, stating it is not applicable constitutes an administrative change.

A3 Unit 1 CTS 3.7.G.6 & 7 and 4.7.G.2 have been deleted. These Specifications were special provisions that expired January 17, 1985, and therefore, no longer apply. As such, the proposed deletion is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 The Surveillance Requirement has been revised to include each manual, power operated, and automatic valve that is not locked, sealed, or otherwise secured in position.

M2 This change adds MODE 2 (STARTUP MODE) to the Applicability to go along with MODE 1 (RUN MODE) which is already required. The CAD System is required to maintain the oxygen concentration in the primary containment below the flammability limit following a LOCA. Adding a new MODE to the Applicability constitutes a more restrictive change. This change is consistent with NUREG-1433.



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.3.1 - CONTAINMENT AIR DILUTION SYSTEM**

- M3 Proposed ACTION C is more restrictive since it requires the unit to be placed in MODE 3 in 12 hours versus CTS 3.7.G.5 which requires that an orderly shutdown be initiated and the reactor to be in the COLD SHUTDOWN CONDITION within 24 hours. In addition, since the existing Specification (CTS 3.7.G.2) is only applicable during the RUN mode (MODE 1), failure to meet the existing specification would only require the unit be placed in at least STARTUP/HOT STANDBY (MODE 2) in 24 hours since at that time CTS 3.7.G.2 is again met.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 This Surveillance is being relocated to plant procedures (IST program) since these valves are tested as part of the IST program. As such, it is not needed to be specified as a specific Surveillance Requirement. If during testing or routine use of the system they are found to be inoperable, the appropriate ACTIONS would be taken. This change is consistent with the BWR Standard Technical Specifications, NUREG 1433.
- LA2 This requirement has been relocated to plant procedures. This type of action is a post-accident action routinely governed by the emergency operating procedures. Any changes to the procedures would be controlled by the licensee controlled programs.

"Specific"

- L1 The Frequency of this Surveillance has been extended to 31 days, similar to other surveillances on tank content (e.g., diesel fuel oil). The nitrogen tank contents only decrease when nitrogen is being added to the drywell, and this evolution is a manually actuated and secured evolution (i.e., it is a very controlled evolution). If nitrogen was being added, it would be monitored more closely. Thus, since there are very positive means to ensure nitrogen tank volume is monitored if being used, and volume does not decrease due to "automatic, unmonitored" use, the 31 day Frequency is considered appropriate.

(A1)

~~3.7.A Primary Containment~~

~~4.7.A Primary Containment~~

3.7.A.4 (Cont'd)

4.7.A.4 (Cont'd)

- c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.
- d. If Specifications 3.7.A.4.a, 3.7.A.4.b, or 3.7.A.4.c cannot be met, the unit shall be placed in a COLD SHUTDOWN CONDITION in an orderly manner within 24 hours.

- c. Each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.22.
- d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

(See Justification for Change For BFN UTS 3.6.17)

~~3.7.A.5 Oxygen Concentration~~

~~4.7.A.5 Oxygen Concentration~~

LC0 3.6.3.2. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

SR 3.6.3.2.1

(LA1)

with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

(L3)

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

(LA4)

Applicability

b. Within the 24-hour period subsequent to placing the reactor in the RUN MODE following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

(L1)

c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a COLD SHUTDOWN CONDITION within 24 hours.

(LA3)

c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

ACTION B

d. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN CONDITION within 24 hours.

Proposed ACTION A

(L2)

BFN Unit 1

3.7/4.7-11

(M1)

AMENDMENT NO 159

PAGE 1 OF 1

~~3.7/A.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

(A1)

~~A.7.A Primary Containment~~

3.7.A.4 (Cont'd)

4.7.A.4 (Cont'd)

- c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.
- d. If Specifications 3.7.A.4.a, .b, or .c cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.

- c. Each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.MM.
- d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

SEE JUSTIFICATION FOR CHANGES FOR BFN ISTS 3.6.1.7

(A1) ~~5. Oxygen Concentration~~

~~5. Oxygen Concentration~~

SR3.6.3.2.1

- LC03.6.3.2 a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen/gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

- a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

Applicability

- b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

- b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

- c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a Cold Shutdown condition within 24 hours.

- c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

- d. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

Propose ACTION A (L2)

ACTION B

BFN Unit 2

3.7/4.7-11

(M1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.A Primary Containment~~

(A1)

~~4.7.A Primary Containment~~

3.7.A.4 (Cont'd)

- c. Two drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening.
- d. If Specifications 3.7.A.4.a, 3.7.A.4.b, or 3.7.A.4.c, cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.

See Justification for Changes for BFNISTS 3.6.1.7

4.7.A.4 (Cont'd)

- c. Once each operating cycle, each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches in accordance with Specification 1.0.MM.
- d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

(A1) ~~5. Oxygen Concentration~~

LCO 3.6.3.2

- a. Containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

(ZA1)

(A2)

Applicability

- b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

(L1)

- c. If plant control air is being used to supply the pneumatic control system inside primary containment, the reactor shall not be started, or if at power, the reactor shall be brought to a Cold Shutdown condition within 24 hours.

(LA3)

- d. If the specifications of 3.7.A.5.a through 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

Action B

(M1)

~~5. Oxygen Concentration~~

SR 3.6.3.2.1

- a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

(L3)

(LA4)

(LA2)

- b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

(LA3)

- c. The control air supply valve for the pneumatic control system inside the primary containment shall be verified closed prior to reactor startup and monthly thereafter.

Proposed Action A

(AL2)

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.3.2
PRIMARY CONTAINMENT OXYGEN CONCENTRATION

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 This statement has been deleted since it is unnecessary. With the reactor in power operation, reactor coolant pressure will always be above 100 psig.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 The requirement to place the plant in Cold Shutdown condition within 24 hours when the limit is not restored within the required Completion Time is revised to reflect placing the plant in a non-applicable condition. CTS 1.0.C.1 states action requirements are applicable during the operational conditions of each specification. Therefore, the requirement to place the plant in Cold Shutdown is not applicable if thermal power is reduced to < 15% RTP (outside the applicable condition) within 8 hours. The current action allows 24 hours to place the plant in a non-applicable condition. As such, this is an additional restriction on plant operation.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 The details of how to reduce oxygen concentration to less than 4% have been eliminated from the ISTS. This type of detail will be retained in plant procedures and/or system operating instructions.
- LA2 Details on the methods of performing surveillances has been relocated to

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.3.2
PRIMARY CONTAINMENT OXYGEN CONCENTRATION

plant procedures. Changes to plant procedures will be controlled by the licensee controlled programs.

- LA3 Requirements for controlling the use of plant control air to supply the pneumatic control system inside the primary containment and the associated surveillance have been relocated to the Technical Requirements Manual (TRM).
- LA4 The requirement to record the containment oxygen concentration will be relocated to plant procedures. Changes to plant procedures will be controlled by the licensee controlled programs.

"Specific"

- L1 The 24 hour allowance for inerting on startup has been changed to allow 24 hours after exceeding 15% power instead of the current Run Mode requirement (approximately 5%). The 24 hour allowance for de-inerting on shutdown has been changed to allow 24 hours prior to reducing below 15% power. These small differences provide some added time to inert or de-inert the drywell, and provide consistency with BWR Standard Technical Specifications, NUREG-1433. These minor changes are justified, since the time allowed without an inerted drywell is only increased slightly, and the fact that at low power levels, hydrogen generation is very small compared to higher power levels.
- L2 Currently, no time is provided to restore oxygen concentration to within limit prior to requiring a plant shutdown. Proposed Required Action A.1 and associated Completion Time will allow 24 hours to restore oxygen to within the limit prior to requiring a plant shutdown. During this time, the CAD System is normally still OPERABLE, thus a means to prevent combustible mixtures still exists. This new ACTION would possibly prevent unnecessary shutdown and the increased potential for transients associated with the shutdown.
- L3 The periodic verification of oxygen concentration in the primary containment has been changed from a daily verification to a weekly verification. The primary containment is inerted to maintain oxygen concentrations within limits. The primary containment leak rate is established for each operating cycle and any changes during normal operation usually occur very slowly. Other changes to primary containment integrity, such as PCIV operability problems, are indicated by other means to the plant operator and appropriate actions are contained in other technical specifications.

UNIT 1

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



MAR 30 1990

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

(A1)

3.7.B. Standby Gas Treatment System

4.7.B. Standby Gas Treatment System

3.7.B.4 (Cont'd)

- b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

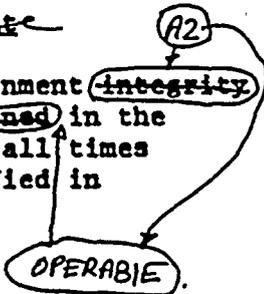
See Justification for Changes for BFN 15TS 3.6.4.7

3.7.C. Secondary Containment

4.7.C. Secondary Containment

- LCD 3.6.4.1 1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.
- Applicability

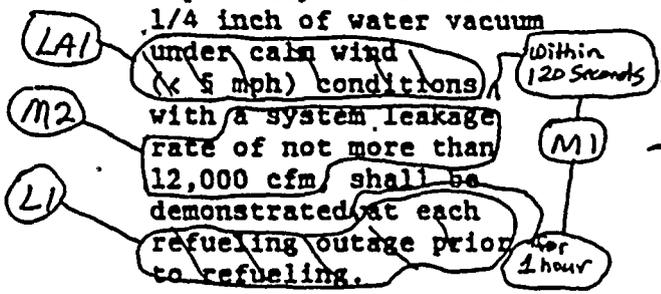
1. Secondary containment surveillance shall be performed as indicated below:



(A1)

SR 3.6.4.1.3 + SR 3.6.4.1.4

- a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system leakage rate of not more than 12,000 cfm shall be demonstrated at each refueling outage prior to refueling.



CONDITION A+C



2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:

- (Proposed Note to Required Action C.1)
- a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel. Immediately M4

ACTION C

- b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

ACTIONS A+B

2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

LA2

M3 Add SRs 3.6.4.1.1 and 3.6.4.1.2



~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.G. Secondary Containment~~

- 3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
- 4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

M5

See Justification for Changes for BFN ISTS 3.6.1.3.

D. Primary Containment Isolation Valves

- 1. When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE* except as specified in 3.7.D.2.

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

D. Primary Containment Isolation Valves

- 1. The primary containment isolation valves surveillance shall be performed as follows:
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation



UNIT 2

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



MAR 30 1990

~~3.7/4.7 CONTAINMENT SYSTEMS~~

LIMITING CONDITIONS FOR OPERATION

A1

~~SURVEILLANCE REQUIREMENTS~~

3.7.B. Standby Gas Treatment System

3.7.B.4 (Cont'd)

b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

4.7.B. Standby Gas Treatment System

See Justification for Changes for BFN 1STS 3.6.4.3

3.7.C. Secondary Containment

LCO 3.6.4.1 Applicability

1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

OPERABLE

4.7.C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

SR 3.6.4.1.3 + SR 3.6.4.1.4

2. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system leakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

LA1

M2

L1

LA2

within 120 seconds

M1

for 1 hour

CONDITION A+C

operable

2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:

LC

Proposed Note to Required Action C.1

ACTION C

a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel immediately

M4

ACTIONS A+B

b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

M3

Add SRs 3.6.4.1.1 and 3.6.4.1.2



a



NOV 16 1992

3.7/4.7 CONTAINMENT SYSTEMS

LIMITING CONDITIONS FOR OPERATION

(A1)

SURVEILLANCE REQUIREMENTS

3.7.6. Secondary Containment

- 3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
- 4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

(M5)

See Justification for Changes for BFN 1STS 3.6.1.3

D. Primary Containment Isolation Valves

- 1. When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE* except as specified in 3.7.D.2.

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

D. Primary Containment Isolation Valves

- 1. The primary containment isolation valves surveillance shall be performed as follows:
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.B. Standby Gas Treatment System

4.7.B. Standby Gas Treatment System

3.7.B.4 (Cont'd)

- b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

See Justification For Changes For GFN 1ST 3.6.4.3

3.7.C. Secondary Containment

4.7.C. Secondary Containment

- 1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

LCO 3.6.4.1 Applicability

* LCO not applicable until just prior to loading fuel into the Unit 3 reactor vessel, provided the Unit 3 reactor zone is not required for secondary containment integrity for other units.

OPERABLE

A3

A2

Condition A+C

Operable

- 2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:

L2

Proposed Note to Revised Action C.1.

- a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel.

Action C

Immediately M4

- b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

Actions A+B

- 1. Secondary containment surveillance shall be performed as indicated below:

SR 3.6.4.1.3 & SR 3.6.4.1.4

- a. Secondary containment capability to maintain 1/4 inch of water vacuum

LA1

under calm wind (< 5 mph) conditions within 120 seconds

M2

with a system inleakage rate of not more than

L1

12,000 cfm, shall be demonstrated at each

LA2

refueling outage prior to refueling.

- 2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

M3

Add SRs 3.6.4.1.1 and 3.6.4.1.2



~~3.7.C. Secondary Containment~~

- 3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.
- 4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:
 - a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.
 - b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones. This is only applicable if reactor zone integrity is required.

(M5)

See Justification for Changes for BFN 1STS 3.6.1.3

D. Primary Containment Isolation Valves

- 1. When Primary Containment Integrity is required, all primary containment isolation valves and all reactor coolant system instrument line flow check valves shall be OPERABLE* except as specified in 3.7.D.2.

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

D. Primary Containment Isolation Valves

- 1. The primary containment isolation valves surveillance shall be performed as follows:
 - a. At least once per operating cycle, the OPERABLE primary containment isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation



**JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.1 - SECONDARY CONTAINMENT**

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The definition of SECONDARY CONTAINMENT INTEGRITY has been deleted from the proposed Technical Specifications. In its place the requirement for secondary containment is that it "shall be OPERABLE." This was done because of the confusion associated with these definitions compared to its use in the respective LCO. The change is editorial in that all the requirements are specifically addressed in the proposed LCO for the secondary containment and in the Secondary Containment Isolation Valves and Standby Gas Treatment System Specifications. The Applicability has been reworded to be consistent with the new definitions of MODES and to have a positive statement as to when it is applicable, not when it is not applicable. Therefore the change is purely a presentation preference adopted by the BWR Standard Technical Specifications, NUREG-1433.

- A3 Amendment 159 to Unit 3 Technical Specifications added a provision to allow separating the Unit 3 reactor zone from the secondary containment envelope under certain conditions (prior to fuel loading) to expedite Unit 3 constructions activities during Unit 2 operation. This provision is no longer needed and can no longer be applied. Therefore the * Note to TS 3.7.C.1 has been deleted. This change is considered administrative since it deletes a requirement that no longer applies.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 This Surveillance (it appears to be only one Surveillance, though it is in two parts) has been broken into two separate Surveillances, SR



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.1 - SECONDARY CONTAINMENT

3.6.1.4.3 and SR 3.6.1.4.4. The tests will ensure the ability of the secondary containment to maintain 1/4 inch vacuum, and in addition, SR 3.6.4.1.3 will ensure the vacuum is attained in 120 seconds, while SR 3.6.4.1.4 will ensure it maintains the vacuum for 1 hour. These new requirements are additional restrictions on plant operation.

- M2 The analysis for secondary containment drawdown assumes two SGT subsystems are needed. Thus, the test now specifies the minimum number of operating SGT subsystems and the total flow rate. To ensure all three SGT subsystems are tested (since the test does not specify that all SGT subsystems must be tested) the Frequency is on a STAGGERED TEST BASIS, which will ensure all three SGT subsystems are tested in 2 cycles. These are additional restrictions of plant operation.
- M3 Two new Surveillance Requirements have been added. SR 3.6.4.1.1 will verify that all secondary containment hatches are closed and sealed every 31 days. SR 3.6.4.1.2 will verify that each access door is closed, except when used for opening, and then one door is closed, every 31 days. These are additional restrictions on plant operation.
- M4 This change requires the movement of irradiated fuel in secondary containment and CORE ALTERATIONS to be "Immediately" suspended if secondary containment is inoperable. In addition, action must be "Immediately" initiated to suspend operations with the potential to drain the reactor vessel in this Condition. The current specification does not establish a time limit to suspend these activities. Immediately suspending these activities minimizes the probability of a fission product release if a reactivity event occurs while the secondary containment is inoperable. Also, immediately initiating action to suspend operation with the potential to drain the reactor vessel will minimize the potential for reactor vessel draindown and subsequent potential for fission release. Imposing a time limit to suspended these activities is a more restrictive change.
- M5 The reactor building is divided into four ventilation zones which may be isolated independently of each other. The refueling room which is common to all three units forms the refueling zone. The individual units below the refueling floor form the other three reactor zones. The zone system is not an engineered safeguard, and the failure of the zone system would not in any way prevent isolation or reduce the capacity of the Secondary Containment System. If the internal zone boundaries should fail, the entire reactor building still meets the requirements of secondary containment. CTS 3.7.C requires the secondary containment integrity to be maintained in the reactor zone and refueling zone at all times except as specified in 3.7.C.2 and 3.7.C.4 respectively. If secondary containment cannot be maintained in the reactor zone, fuel



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.1 - SECONDARY CONTAINMENT

secondary containment must be restored within 4 hours or all reactor shall be shut down. If secondary containment cannot be maintained in the refueling zone, the handling of spent fuel and all operations over spent fuel pools and open reactor wells shall be prohibited.

Currently, a combined secondary containment integrity test is performed to demonstrate Technical Specification operability. In addition, due to leakage between zones, zone integrity is difficult to maintain. As such, secondary containment integrity is maintained on the three reactor zones and the refueling zone at all time. Therefore, the separate Specification that only prohibits the handling of spent fuel and all operations over spent fuel pools and open reactor wells when refueling zone integrity is not maintained is not necessary and has been deleted

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 This design detail/requirement has been relocated to the Background section of the Bases for ITS 3.6.4.3, "Standby Gas Treatment System," and to plant procedures governing this Surveillance Requirement. Any changes to this requirement will require a licensee controlled program evaluation.
- LA2 The requirement to operate the Standby Gas Treatment System after a secondary containment violation is determined and has been isolated (i.e., restored) to check if it can maintain the proper vacuum is being relocated to plant procedures. Any time the OPERABILITY of a system or component has been affected by maintenance, replacement, or repair, post maintenance testing is required to demonstrate OPERABILITY of the system or components. Explicit post maintenance surveillance testing has therefore been deleted from the Technical Specifications and will be relocated to the appropriate plant procedures. Any changes to the requirement will require a licensee controlled program evaluation. This change is consistent with NUREG-1433.

"Specific"

- L1 The proposed surveillances for the 1/4 inch vacuum tests do not include the restriction on plant conditions that requires the surveillances to be performed during a refueling outage, prior to refueling. These Surveillances could be adequately performed in other than a refueling outage without jeopardizing safe plant operations. The control of the plant conditions appropriate to perform the test is an issue for procedures and scheduling, and has been determined by the NRC Staff to

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.1 - SECONDARY CONTAINMENT

plant conditions appropriate to perform the test is an issue for procedures and scheduling, and has been determined by the NRC Staff to be unnecessary as a Technical Specification restriction. As indicated in Generic Letter 91-04, allowing this control is consistent with the vast majority of other Technical Specification surveillances that do not dictate plant conditions for the surveillances. The proposed change to the 18 month frequency also effectively increases the surveillance interval. The current Technical Specification for all three units requires performance at each refueling outage prior to refueling. Since the secondary containment is common to all three BFN units, with all three units operating, this could result in performance of the same test at an average of every 6 months. The change to the 18 month frequency will allow this test to be performed once and applied to all three units Technical Specifications. Since operating experience has shown these component usually pass the Surveillance at the 18 month frequency, the frequency is considered acceptable from a reliability standpoint.

- L2 Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operation and the inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown. By adding an exception to LCO 3.0.3 for the failing to suspend irradiated fuel movement, an LCO 3.0.3 required reactor shutdown is avoided in MODE 1, 2, or 3. However, the plant would still be required to shutdown after 4 hours per proposed Required Actions B.1 and B.2 in addition to suspending fuel movement per Required Action C.1.



UNIT 1
CURRENT
TECHNICAL
SPECIFICATION
MARKUP



MAR 30 1990

~~LIMITING CONDITIONS FOR OPERATION~~ (A)

SURVEILLANCE REQUIREMENTS

3.7.B. Standby Gas Treatment System

4.7.B. Standby Gas Treatment System

3.7.B.4 (Cont'd)

- b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

See Justification for Changes for BFN 1STS 3.6.4.3

(A1)

~~3.7.C. Secondary Containment~~

4.7.C. Secondary Containment

(A2)

Proposed LCO 3.6.4.2 + Applicability

- 1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

- 1. Secondary containment surveillance shall be performed as indicated below:

(A3)

Proposed Notes 2+3 to actions

- a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system leakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

Proposed Note 1 to Actions

(L1)

- 2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:

- 2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

(L2)

Proposed Note to Required Action D.1

(ACTION D)

- a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel. *immediately* (M2)

(L1)

(ACTIONS A+B)

- b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

(ACTION C)

See Justification for changes for BFN 1STS 3.6.4.1

(M1)

Proposed SRs 3.6.4.2.1, 3.6.4.2.2

UNIT 2

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

MAR 30 1990

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

3.7.B. Standby Gas Treatment System

3.7.B.4 (Cont'd)

- b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

4.7.B. Standby Gas Treatment System

See Justification for Changes for BFN 1STS 3.6.4.3

3.7.C. Secondary Containment (A1)

- 1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

4.7.C. Secondary Containment

- 1. Secondary containment surveillance shall be performed as indicated below:
 - a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system leakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.
- 2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

See Justification for Changes for BFN 1STS 3.6.4.1

Proposed SRS 3.6.4.2.1, 3.6.4.2.2

(M1)

(A2)

Proposed LCO 3.6.4.2 Applicability

(A3)

Proposed Notes 2+3 to ACTIONS

(L1)

Proposed Note 1 to ACTIONS

(L2)

Proposed Note to Required Action D.1

(ACTION D)

- 2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:
 - a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel. *immediately* (M1)
 - b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

(L1) ACTIONS A+B

(ACTION C)

UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP

NOV 18 1991

A1

3.7.B. Standby Gas Treatment System

3.7.B.4 (Cont'd)

- b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

4.7.B. Standby Gas Treatment System

see justification for changes for BFN ISTS 3.6.4.3

3.7.C. Secondary Containment

Proposed LCO 3.6.4.2 + Applicability

- 1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

* LCO not applicable until just prior to loading fuel into the Unit 3 reactor vessel, provided the Unit 3 reactor zone is not required for secondary containment integrity for other units.

Proposed Notes 2 + 3 to Actions

Proposed Note 1 to Actions

- 2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:

Proposed Note to Required Action D.1

- a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel. *immediately*

- b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

4.7.C. Secondary Containment

- 1. Secondary containment surveillance shall be performed as indicated below:
 - a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system inleakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.
- 2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

see justification for changes for BFN ISTS 3.6.4.1

Proposed SRs 3.6.4.2.1, 3.6.4.2.2



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.2
SECONDARY CONTAINMENT ISOLATION VALVES

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The current definition of Secondary Containment Integrity requires all secondary containment isolation valves (SCIVs) to be OPERABLE or in their isolation position. Thus, the current secondary containment Specification encompasses the SCIV requirements. It is proposed to provide a separate Specification for SCIVs for clarity. Thus, the new LCO will require all SCIVs to be OPERABLE, consistent with the current requirements. The applicability has been reworded to be consistent with the new definitions of MODES and to have a positive statement as to when it is applicable, not when it is not applicable.

- A3 Proposed ACTIONS Note 2 ("Separate Condition entry is allowed for each penetration flow path") provides explicit instructions for proper application of the ACTIONS for Technical Specification compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," this Note provides direction consistent with the intent of the existing Actions for inoperable isolation valves. Similarly, proposed ACTIONS Note 3 facilitates the use and understanding of the intent to consider the operability of any system affected by inoperable isolation valves and to apply applicable Actions. With the proposed LCO 3.0.6, this intent would not necessarily apply. This clarification is consistent with the intent and interpretation of the existing Technical Specifications, and is therefore considered an administrative presentation preference.



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.2
SECONDARY CONTAINMENT ISOLATION VALVES

- A4 Amendment 159 to Unit 3 Technical Specifications added a provision to allow separating the Unit 3 reactor zone from the secondary containment envelope under certain conditions (prior to fuel loading) to expedite Unit 3 construction activities during Unit 2 operation. This provision is no longer needed and can no longer be applied. Therefore the * Note to TS 3.7.C.1 has been deleted. This change is considered administrative since it deletes a requirement that no longer applies.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 Two new Surveillance Requirements have been added to ensure SCIV operability. SR 3.6.4.2.1 verifies that SCIVs isolate within the assumed times in accordance with the inservice testing program. SR 3.6.4.2.2 verifies that each SCIV actuates to its isolation position on an accident signal every 18 months. These are additional restrictions on plant operation.
- M2 This change requires the movement of irradiated fuel in secondary containment and CORE ALTERATIONS to be "immediately" suspended if secondary containment is inoperable. In addition, action must be "immediately" initiated to suspend operations with the potential to drain the reactor vessel in this Condition. The current specification does not establish a time limit to suspend these activities. Immediately suspending these activities minimizes the probability of a fission product release if a reactivity event occurs while the secondary containment is inoperable. Also, immediately initiating action to suspend operation with the potential to drain the reactor vessel will minimize the potential for reactor vessel draindown and subsequent potential for fission release. Imposing a time limit to suspended these activities is a more restrictive change.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Specific"

- L1 This Action has been changed to allow one valve in a penetration to be inoperable for up to 8 hours, instead of the current 4 hours. Proposed ACTION A now requires the penetration to be isolated in 8 hours. This is justified since an OPERABLE valve in the penetration remains to isolate the penetration if needed, thus the "leak tightness" of the secondary containment is still maintained. The isolated penetration is required to be verified every 31 days while a valve is inoperable, further ensuring the continued "leak tightness" of the secondary containment. Proposed ACTION B will verify that if both SCIVs in a penetration are inoperable, at least one SCIV in a penetration is closed

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.2
SECONDARY CONTAINMENT ISOLATION VALVES

within 4 hours. This maintains consistency with the current requirements. An allowance is proposed for intermittently opening closed secondary containment isolation valves under administrative control. The allowance is presented in proposed ACTIONS Note 1, which allows opening of secondary containment penetrations on an intermittent basis for performing Surveillances, repairs, routine evolutions, etc.

- L2 Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operation and the inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown. By adding an exception to LCO 3.0.3 for the failing to suspend irradiated fuel movement, an LCO 3.0.3 required reactor shutdown is avoided in MODE 1, 2, or 3. However, the plant would still be required to shutdown per proposed Required Actions C.1 and C.2 in addition to suspending fuel movement per Required Action D.1. However, this shutdown is considered less restrictive since Required Action C.1 allows the plant to be in Hot Shutdown within 12 hours versus Hot Standby within 6 hours as required by CTS 1.0.C.1. Both CTS and the proposed Required Action C.2 require the plant to be in Cold Shutdown within 36 hours.



UNIT 1

**CURRENT
TECHNICAL
SPECIFICATION**

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FEB 13 1995

~~3.7/4.7 CONTAINMENT SYSTEMS~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B. Standby Gas Treatment System~~

~~4.7.B. Standby Gas Treatment System~~

2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at $\geq 10\%$ design flow on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

b. The results of laboratory carbon sample analysis shall show $\geq 90\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.

c. System shall be shown to operate within $\pm 10\%$ design flow.

(A2)

2. a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once per operating cycle or once every 18 months whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.



MAR 30 1990

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B Standby Gas Treatment System~~

~~4.7.B Standby Gas Treatment System~~

(A4) Proposed ACTION D →

(A3) ON an actual or simulated initiation signal

ACTION A
3. From and after the date that one train of the standby gas treatment system is made or found to be inoperable for any reason, REACTOR POWER OPERATION and fuel handling is permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE, provided that during such 7 days all active components of the other two standby gas treatment trains shall be operable.

(L3) Proposed Note to Required Actions of C+E

(L2) Proposed Required Action C

ACTIONS C+E
4. If these conditions cannot be met:
a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel.

(M1) Immediately

~~4.7.B.2 (Cont'd)~~

SR 3.6.4.3.1

(M2) d. Each train shall be operated a total of at least 10 hours every month.

(LA1)

e. Test sealing of gaskets for housing doors shall be performed utilizing chemical smoke generators during each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a

(A5)

SR 3.6.4.3.2

3. a. Once per operating cycle, automatic initiation of each branch of the standby gas treatment system shall be demonstrated from each unit's controls.

(18 months)

(LA2)

SR 3.6.4.3.4

b. At least once per year manual operability of the bypass valve for filter cooling shall be demonstrated.

(12 months)

(A1)

(L1)

c. When one train of the standby gas treatment system becomes inoperable the other two trains shall be demonstrated to be OPERABLE within 2 hours and daily thereafter.

(A1)

during movement of irradiated fuel assemblies in the secondary containment during CORE ALTERATION, or during OPRDRs

MAR 30 1990

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B. Standby Gas Treatment System~~ (A1)

~~4.7.B. Standby Gas Treatment System~~

~~3.7.B.4 (Cont'd)~~

ACTION
B

b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

see Justification for Change for BFN ISTS 3.6.4.1

3.7.C. Secondary Containment

4.7.C. Secondary Containment

- 1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

- 1. Secondary containment surveillance shall be performed as indicated below:

- a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system leakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

- 2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:
 - a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel.
 - b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

- 2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.



~~3.7/4.7 CONTAINMENT SYSTEMS~~

Specification 3.6.4.3
MAR 30 1990

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B. Standby Gas Treatment System~~

~~4.7.B. Standby Gas Treatment System~~

1. Except as specified in Specification 3.7.B.3 below, all three trains of the standby gas treatment system shall be OPERABLE at all times when secondary containment integrity is required.

LC 3.6.4.3

Applicability

at all times when secondary containment integrity is required.

See Justification for Change for BFN ISTS Section 5.0

- 1. At least once per year, the following conditions shall be demonstrated.
 - a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at a flow of 9000 cfm ($\pm 10\%$).
 - b. The inlet heaters on each circuit are tested in accordance with ANSI N510-1975, and are capable of an output of at least 40 kW.
 - c. Air distribution is uniform within 20% across HEPA filters and charcoal adsorbers.

A2

Proposed SR 3.6.4.3.2



(A1)

3.7.F. Primary Containment Purge System

4.7.F. Primary Containment Purge System

1. The primary containment purge system shall be OPERABLE for PURGING, except as specified in 3.7.F.2.
 - a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show \geq 99% DOP removal and \geq 99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.
 - b. The results of laboratory carbon sample analysis shall show \geq 85% radioactive methyl iodide removal when tested in accordance with ASTM D3803.
 - c. System flow rate shall be shown to be within \pm 10% of design flow when tested in accordance with ANSI N510-1975.
2. If the provisions of 3.7.F.1.a, b, and c cannot be met, the system shall be declared inoperable. The provisions of Technical Specification 1.C.1 do not apply. PURGING may continue using the Standby Gas Treatment System.

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8.5 inches of water at system design flow rate (\pm 10%).
 - a. The tests and sample analysis of Specification 3.7.F.1 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first or after 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.
 - b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
 - c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

3. a. The 18-inch primary containment isolation valves associated with PURGING may be open during the RUN mode for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. The OPERABILITY of

See Justification for Changes for BFN 15TS 3.6.1.3

(R1)

UNIT 2

**CURRENT
TECHNICAL
SPECIFICATION**

MARKUP



~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B. Standby Gas Treatment System~~

(A1)

~~4.7.B. Standby Gas Treatment System~~

2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at $\geq 10\%$ design flow on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

b. The results of laboratory carbon sample analysis shall show $\geq 90\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.

c. System shall be shown to operate within $\pm 10\%$ design flow.

(A2)

2. a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once per operating cycle or once every 18 months whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

MAR 30 1990

~~3.7/A.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B. Standby Gas Treatment System~~ (A1)

~~4.7.B. Standby Gas Treatment System~~

(A4) Proposed ACTION D →

~~4.7.B.2 (Cont'd)~~

~~SR 3.6.4.3.1~~

d. Each train shall be operated a total of at least 10 hours every month. (M2)

e. Test sealing of gaskets for housing doors shall be performed utilizing chemical smoke generators during each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a. (LA1)

~~SR 3.6.4.3.3~~

3. a. Once per operating cycle ^{18 months} automatic initiation of each branch of the standby gas treatment system shall be demonstrated from each unit's controls. (LA2)

~~SR 3.6.4.3.4~~

b. At least ^{12 months} once per year manual operability of the bypass valve for filter cooling shall be demonstrated. (A1)

c. When one train of the standby gas treatment system becomes inoperable the other two trains shall be demonstrated to be OPERABLE within 2 hours and daily thereafter. (L1)

(A1) during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPERATIONS (M1)

3. From and after the date that one train of the standby gas treatment system is made or found to be inoperable for any reason, REACTOR POWER OPERATION and fuel handling is permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE, provided that during such 7 days all active components of the other two standby gas treatment trains shall be operable.

ACTION A

(L3) Proposed Notes to Required Actions of C+E:1

(L2) Proposed Required Action C:1 →

4. If these conditions cannot be met:

a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel immediately (M1)

ACTIONS C+E

MAR 30 1990

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B. Standby Gas Treatment System~~

~~4.7.B. Standby Gas Treatment System~~

(A1)

~~3.7.B.4 (Cont'd)~~

ACTION B

- b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

See Justification for Changes for BFN ISTS 3.6.4.1

3.7.C. Secondary Containment

4.7.C. Secondary Containment

1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.
2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:
 - a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel.
 - b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

1. Secondary containment surveillance shall be performed as indicated below:
 - a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system leakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.
2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.



MAR 30 1990

~~3.7/4.7 CONTAINMENT SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~

(A1)

~~SURVEILLANCE REQUIREMENTS~~

~~3.7.B. Standby Gas Treatment System~~

- 1. Except as specified in Specification 3.7.B.3 below, all three trains of the standby gas treatment system shall be OPERABLE at all times when secondary containment integrity is required.

LC0 36.4.3

Applicability

See Justification for Changes for BFN ISTS Section 5.0

~~4.7.B. Standby Gas Treatment System~~

- 1. At least once per year, the following conditions shall be demonstrated.
 - a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at a flow of 9000 cfm ($\pm 10\%$).
 - b. The inlet heaters on each circuit are tested in accordance with ANSI N510-1975, and are capable of an output of at least 40 kW.
 - c. Air distribution is uniform within 20% across HEPA filters and charcoal adsorbers.

(A2)

Proposal SR 3.6.4.3.2



(A1)

FEB 13 1995

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.F. Primary Containment Purge System

1. The primary containment purge system shall be OPERABLE for PURGING, except as specified in 3.7.F.2.
 - a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.
 - b. The results of laboratory carbon sample analysis shall show $\geq 85\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.
 - c. System flow rate shall be shown to be within $\pm 10\%$ of design flow when tested in accordance with ANSI N510-1975.
2. If the provisions of 3.7.F.1.a, b, and c cannot be met, the system shall be declared inoperable. The provisions of Technical Specification 1.C.1 do not apply. PURGING may continue using the Standby Gas Treatment System.

RI

T

3. a. The 18-inch primary containment isolation valves associated with PURGING may be open during the RUN mode for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. The OPERABILITY of

4.7.F. Primary Containment Purge System

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8.5 inches of water at system design flow rate ($\pm 10\%$).
 - a. The tests and sample analysis of Specification 3.7.F.1 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first or after 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.
 - b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
 - c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

See Justification for Changes for BFN 1STS 3.6.1.3



UNIT 3

CURRENT
TECHNICAL
SPECIFICATION

MARKUP



(A1)

~~3.7.B. Standby Gas Treatment System~~

1. Except as specified in Specification 3.7.B.3 below, all three trains of the standby gas treatment system shall be OPERABLE at all times when secondary containment integrity is required.

LCD 3.6.4.3

Applicability

See Justification for Changes for BFN ISTS Section 5.0

~~4.7.B. Standby Gas Treatment System~~

1. At least once per year, the following conditions shall be demonstrated.

- a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at a flow of 9000 cfm ($\pm 10\%$).
- b. The inlet heaters on each circuit are tested in accordance with ANSI N510-1975, and are capable of an output of at least 40 kW.
- c. Air distribution is uniform within 20% across HEPA filters and charcoal adsorbers.

(A2) Proposed SR 3.6.4.3.2



(A1)

~~3.7.B. Standby Gas Treatment System~~

~~4.7.B. Standby Gas Treatment System~~

2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at $\geq 10\%$ design flow on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

b. The results of laboratory carbon sample analysis shall show $\geq 90\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.

c. System shall be shown to operate within $\pm 10\%$ design flow.

2. a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once per operating cycle or once every 18 months whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

F

(A2)

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7.B. Standby Gas Treatment System

4.7.B. Standby Gas Treatment System

Proposed Action D →

(A4)

(A3)

on an actual or simulated initiation signal

Action A

3. From and after the date that one train of the standby gas treatment system is made or found to be inoperable for any reason, REACTOR POWER OPERATION and fuel handling is permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE, provided that during such 7 days all active components of the other two standby gas treatment trains shall be operable.

(L3) Proposed Note to Required Actions of C+E.1

(L2) Proposed Required Action C.1 →

4. If these conditions cannot be met:

Actions C+E

a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel

immediately (M1)

4.7.B.2 (Cont'd)

SR 3.6.4.3.1

(M2)

d. Each train shall be operated a total of at least 10 hours every month.

e. Test sealing of gaskets for housing doors shall be performed utilizing chemical smoke generators during each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a.

(LA1)

SR 3.6.4.3.2

(A5) 12 months

3. a. Once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated from each unit's controls

(LA2)

SR 3.6.4.3.4

(A1) 12 months

b. At least once per year manual operability of the bypass valve for filter cooling shall be demonstrated.

c. When one train of the standby gas treatment system becomes inoperable the other two trains shall be demonstrated to be OPERABLE within 2 hours and daily thereafter.

(L1)

(A1)

during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPRVs



(A1)

~~3.7.B. Standby Gas Treatment System~~

~~4.7.B. Standby Gas Treatment System~~

~~3.7.B.4 (Cont'd)~~

Action B

- b. Place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

See Justification for Changes for BFN ISTS 3.6.4.1

3.7.C. Secondary Containment

4.7.C. Secondary Containment

- * 1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.
- * LCO not applicable until just prior to loading fuel into the Unit 3 reactor vessel, provided the Unit 3 reactor zone is not required for secondary containment integrity for other units.
- 2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:
 - a. Suspend all fuel handling operations, core alterations, and activities with the potential to drain any reactor vessel containing fuel.
 - b. Restore reactor zone secondary containment integrity within 4 hours, or place all reactors in at least a HOT SHUTDOWN CONDITION within the next 12 hours and in a COLD SHUTDOWN CONDITION within the following 24 hours.

- 1. Secondary containment surveillance shall be performed as indicated below:
 - a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system inleakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.
- 2. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.



(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~SURVEILLANCE REQUIREMENTS~~

3.7.F. Primary Containment Purge System

4.7.F. Primary Containment Purge System

1. The primary containment purge system shall be OPERABLE for PURGING, except as specified in 3.7.F.2.
 - a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.
 - b. The results of laboratory carbon sample analysis shall show $\geq 85\%$ radioactive methyl iodide removal when tested in accordance with ASTM D3803.
 - c. System flow rate shall be shown to be within $\pm 10\%$ of design flow when tested in accordance with ANSI N510-1975.
2. If the provisions of 3.7.F.1.a, b, and c cannot be met, the system shall be declared inoperable. The provisions of Technical Specification 1.C.1 do not apply. PURGING may continue using the Standby Gas Treatment System.

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 8.5 inches of water at system design flow rate ($\pm 10\%$).
 - a. The tests and sample analysis of Specification 3.7.F.1 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first or after 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.
 - b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
 - c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

3. a. The 18-inch primary containment isolation valves associated with PURGING may be open during the RUN mode for a 24-hour period after entering the RUN mode and/or for a 24-hour period prior to entering the SHUTDOWN mode. The OPERABILITY of

See justification for changes for BFN 15TS 3.6.1.3

(R1) →



JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.3 - STANDBY GAS TREATMENT SYSTEM

ADMINISTRATIVE CHANGES

- A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

- A2 The technical content of this requirement is being moved to Section 5.0 of the proposed Technical Specifications in accordance with the format of the BWR Standard Technical Specifications, NUREG 1433. Any technical changes to this requirement will be addressed within the content of proposed Specification 5.5.7. A surveillance requirement (proposed SR 3.6.4.3.2) is added to clarify that the tests of the Ventilation Filter Testing Program must also be completed and passed for determining operability of the SGT System. Since this is a presentation preference that maintains current requirements, this change is considered administrative.
- A3 The description of the signal used to automatically initiate the SGT System "actual or simulated initiation signal" has been added for clarity. This is consistent with the BWR Standard Technical Specifications, NUREG 1433, and no change is intended.
- A4 A new ACTION is proposed (ACTION D) which directs entry into LCO 3.0.3 if two or more required standby gas treatment subsystems are inoperable in Modes 1, 2, or 3. This avoids confusion as to the proper action if in Modes 1, 2, or 3 and simultaneously handling fuel, conducting CORE ALTERATIONS, or operations with the potential for draining the reactor vessel. Since the proposed ACTION effectively results in the same action as the current specification, this change is considered administrative.
- A5 The Frequency for verifying SGTS automatic initiation has been changed to 18 months from once per operating cycle. The BFN operating cycle is currently defined as 18 months. As such this is a change in presentation only and is therefore administrative.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.3 - STANDBY GAS TREATMENT SYSTEM

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 This change requires the movement of irradiated fuel in secondary containment and CORE ALTERATIONS to be "Immediately" suspended if secondary containment is inoperable. In addition, action must be "Immediately" initiated to suspend operations with the potential to drain the reactor vessel in this Condition. The current specification does not establish a time limit to suspend these activities. Immediately suspending these activities minimizes the probability of a fission product release if a reactivity event occurs while the secondary containment is inoperable. Also, immediately initiating action to suspend operation with the potential to drain the reactor vessel will minimize the potential for reactor vessel draindown and subsequent potential for fission release. Imposing a time limit to suspended these activities is a more restrictive change.
- M2 CTS 4.7.B.2.d requires each train to be operated a total of at least 10 hours each month. Proposed SR 3.6.4.3.1 requires each train to be operated continuously for 10 hours. As such, the proposed SR is considered more restrictive.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA1 Details on methods of testing gasket seals for housing doors has been deleted. This type of detail will be retained in plant procedures and/or system operating instructions.
- LA2 Details on the method of performing Standby Gas Treatment system surveillance requirements have been relocated to plant procedures. Changes to the procedure will be controlled by licensee controlled programs.

"Specific"

- L1 The proposed change will delete the requirement to test the other SGT subsystems when one subsystem is inoperable. The requirement for demonstrating operability of the redundant subsystems was originally chosen because there was a lack of plant operating history and a lack of sufficient equipment failure data. Since that time, plant operating experience has demonstrated that testing of the redundant subsystems when one subsystem is inoperable is not necessary to provide adequate assurance of system operability.

This change will allow credit to be taken for normal periodic surveillances as a demonstration of operability and availability of the

**JUSTIFICATION FOR CHANGES
BFN ISTS 3.6.4.3 - STANDBY GAS TREATMENT SYSTEM**

remaining components. The periodic frequencies specified to demonstrate operability of the remaining components have been shown to be adequate to ensure equipment operability. As stated in NRC Generic Letter 87-09, "It is overly conservative to assume that systems or components are inoperable when a surveillance requirement has not been performed. The opposite is in fact the case; the vast majority of surveillances demonstrate the systems or components in fact are operable." Therefore, reliance on the specified surveillance intervals does not result in a reduced level of confidence concerning the equipment availability. Also, the current Standard Technical Specifications (STS) and, more specifically, all the Technical Specifications approved for recently licensed BWRs accept the philosophy of system operability based on satisfactory performance of monthly, quarterly, refueling interval, post maintenance or other specified performance tests without requiring additional testing when another system is inoperable (except for diesel generator testing, which is not being changed).

- L2 An alternative is proposed to suspending operations if a SGT subsystem cannot be returned to OPERABLE status within seven days, and movement of irradiated fuel assemblies, CORE ALTERATIONS, or operations with the potential for draining the reactor vessel are being conducted. The alternative is to initiate two OPERABLE subsystems of SGT and continue to conduct the operations. Since two subsystems are sufficient for any accident, the risk of failure of the subsystems to perform their intended function is significantly reduced if they are running. This alternative is less restrictive than the existing requirement. However, the proposed alternative ensures that the remaining subsystems are Operable, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected. This change is consistent with NUREG-1433.
- L3 The Required Actions of C and E.1 have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operation and the inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown. By adding an exception to LCO 3.0.3 for the failing to suspend irradiated fuel movement, an LCO 3.0.3 required reactor shutdown is avoided in MODE 1, 2, or 3. However, the plant would still be required to shutdown per proposed Required Actions B.1 and B.2 in addition to suspending fuel movement per Required Actions C.1 and E.1. However, this shutdown is considered less restrictive since Required Action B.1 allows the plant to be in Hot Shutdown within 12 hours versus Hot Standby within 6 hours as required by CTS 1.0.C.1. Both CTS and the proposed Required Action B.2 require the plant to be in Cold Shutdown within 36 hours.



JUSTIFICATION FOR CHANGES
CTS 3.7.F/4.7.F - PRIMARY CONTAINMENT PURGE SYSTEM

RELOCATED CHANGES

- R1 CTS 3.7.F.1 & 2 and 4.7.F requirements have been relocated to the Technical Requirements Manual (TRM). The Primary Containment Purge System is normally isolated and normally not required to be functional during power operation. It does provide the preferred exhaust path for purging the primary containment; however, the SGTS can be used to perform the equivalent function. The supply and isolation valves are depended on to function properly for containment isolation, which is covered in proposed BFN ISTS Section 3.6.1.3, Primary Containment Isolation Valves.

