Charles H. Cruse
Vice President
Nuclear Energy

1650 Calvert Cliffs Parkway
Lusby, Maryland 20657
410 495-4455

Constellation Nuclear

Calvert Cliffs
Nuclear Power Plant

A Member of the Constellation Energy Group

November 13, 2001

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Technical Specification Bases, Revisions 10, 11, and 12

Enclosed for your use is one copy each of the Calvert Cliffs Technical Specifications Bases, Revisions 10, 11, and 12. These revisions were performed under the Technical Specification Bases Control Program (Technical Specification 5.5.14). This program states, "Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e)." Revisions 10 and 11 were distributed to the Calvert Cliffs site earlier in 2001.

The List of Effective pages is included. Please replace the appropriate pages of your copies of the Technical Specification Bases with these enclosed pages.

Should you have questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,

CHC/PSF/bjd

Enclosures: As stated

cc: **(Without Enclosures)**
R. S. Fleishman, Esquire
J. E. Silberg, Esquire
Director, Project Directorate I-1, NRC
D. M. Skay, NRC
H. J. Miller, NRC
Resident Inspector, NRC
R. I. McLean, DNR

A001

PAGE REPLACEMENT INSTRUCTIONS
Calvert Cliffs Nuclear Power Plant

Technical Specification Bases - Revision 10

Remove and Discard	**Insert**
List of Revisions	
LOR-1	LOR-1
List of Effective Pages	
LEP-1 through LEP-5	LEP-1 through LEP-5
Technical Specification Bases Pages	
B 3.4.13-1 B 3.4.13-2	B 3.4.13-1 * B 3.4.13-2
B 3.5.1-7 B 3.5.1-8 B 3.5.1-9	B 3.5.1-7 * B 3.5.1-8 B 3.5.1-9 **
B 3.7.1-3 B 3.7.1-4	B 3.7.1-3 * B 3.7.1-4
B 3.8.1-3 B 3.8.1-4	B 3.8.1-3 * B 3.8.1-4
B 3.8.2-3 B 3.8.2-4	B 3.8.2-3 B 3.8.2-4 *
B 3.9.1-1 B 3.9.1-2 B 3.9.1-3 B 3.9.1-4	B 3.9.1-1 * B 3.9.1-2 B 3.9.1-3 ** B 3.9.1-4 **

* Unchanged, backside of page
** Rollover page

TECHNICAL SPECIFICATION BASES

LIST OF REVISIONS AND ISSUE DATES

Rev.	Date Issued	Date to NRC
0		May 4, 1998
1	August 28, 1998	October 30, 1998
2	August 28, 1998	October 30, 1998
3	October 28, 1998	October 30, 1998
4	March 16, 1999	October 18, 1999
5	October 18, 1999	October 18, 1999
6	April 14, 2000	October 24, 2000
7	May 18, 2000	October 24, 2000
8	June 29, 2000	October 24, 2000
9	October 24, 2000	October 24, 2000
10	February 1, 2001	

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core makeup the RCS. Component joints are made by welding, bolting, rolling, or pressure loading. Valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

Reference 1, Appendix 1C, Criterion 16 requires means for detecting reactor coolant LEAKAGE. Reference 2 describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the facility and the public.

A limited amount of leakage inside Containment Structure is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS LEAKAGE detection.

This LCO deals with protection of the RCPB from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a LOCA.

BASES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 100 gpd/SG primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside the Containment Structure resulting from a steam line break accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a SGTR. The leakage contaminates the secondary fluid.

Reference 1, Section 14.15 analysis for SGTR assumes the contaminated secondary fluid is released via the atmospheric dump valves and main steam safety valves. Most of the released radiation is due to the ruptured tube. The 100 gpd/SG primary to secondary LEAKAGE is relatively inconsequential.

The steam line break is more limiting for site radiation releases. The safety analysis for the steam line break accident assumes 100 gpd/SG primary to secondary LEAKAGE as an initial condition. The dose consequences resulting from the steam line break accident are described in Reference 1, Section 14.14.

Reactor Coolant System operational LEAKAGE satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

Reactor Coolant System operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

C.1 and C.2

If the SIT cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and MODE 4 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.

D.1

If more than one SIT is inoperable, the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification every 12 hours that each SIT isolation valve is fully open, as indicated in the Control Room, ensures that SITs are available for injection and ensures timely discovery if a valve should be partially closed. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor-operated valve should not change position with power removed, a closed valve could result in not meeting accident analysis assumptions. A 12 hour Frequency is considered reasonable in view of other administrative controls that ensure the unlikelihood of a mispositioned isolation valve.

SR 3.5.1.2 and SR 3.5.1.3

Safety injection tank borated water volume and nitrogen cover pressure should be verified to be within specified limits every 12 hours in order to ensure adequate injection during a LOCA. Due to the static design of the SITs, a 12 hour Frequency usually allows the operator sufficient time to identify changes before the limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

Thirty-one days is reasonable for verification to determine that each SIT's boron concentration is within the required limits, because the static design of the SITs limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms, such as stratification or inleakage.

Verification consists of monitoring inleakage or sampling. All intentional sources of level increase are maintained administratively to ensure SIT boron concentrations are within technical specification limits. A sample of the SIT is required, to verify boron concentration, if 10 inches or greater of inleakage has occurred since last sampled.

Sampling the affected SIT (by taking the sample at the discharge of the operating HPSI pump) within one hour prior to a 1% volume increase of normal tank volume, will ensure the boron concentration of the fluid to be added to the SIT is within the required limit prior to adding inventory to the SIT(s).

SR 3.5.1.5

Verification every 31 days that power is removed from each SIT isolation valve operator, by maintaining the feeder breaker open under administrative control, when the pressurizer pressure is ≥ 2000 psig ensures that an active failure could not result in the undetected closure of an SIT motor-operated isolation valve. If this were to occur, only two SITs would be available for injection, given a single failure coincident with a LOCA. Since installation and removal of power to the SIT isolation valve operators is conducted under administrative control, the 31 day Frequency was chosen to provide additional assurance that power is removed.

This SR allows power to be supplied to the motor-operated isolation valves when RCS pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during unit startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure

BASES

interlock associated with the valves. Should closure of a valve occur in spite of the interlock, the safety injection signal provided to the valves would open a closed valve in the event of a LOCA.

REFERENCES

1. Institute of Electrical and Electronic Engineers Standard 279-1971, "IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations"
2. Updated Final Safety Analysis Report (UFSAR)
3. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

BASES

This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences of accidents that could result in a challenge to the reactor coolant pressure boundary.

APPLICABILITY

In MODEs 1, 2, and 3, a minimum of five MSSVs per steam generator are required to be OPERABLE, according to Table 3.7.1-1 in the accompanying LCO, which is limiting and bounds all lower MODEs.

In MODEs 4 and 5, there are no credible transients requiring the MSSVs.

The steam generators are not normally used for heat removal in MODEs 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODEs.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets Code requirements for the power level. The number of inoperable MSSVs will determine the necessary level of reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the power level-high channels. The reactor trip setpoint reductions are derived on the following basis:

$$SP = \frac{(X) - (Y)(V)}{X} \times 106.5$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER
V = maximum number of inoperable safety valves per steam line
106.5 = Power Level-High Trip Setpoint
X = Total relieving capacity of all safety valves per steam line in lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour

Nuclear Regulatory Commission Information Notice 94-60 states that the linear relationship is not always valid; however, the setpoints in Table 3.7.1-1 have been verified by transient analyses.

The operator should limit the maximum steady state power level to some value slightly below this setpoint to avoid an inadvertent overpower trip.

The four hour Completion Time for Required Action A.2 is consistent with A.1. An additional eight hours is allowed to reduce the setpoints in recognition of the difficulty of resetting all channels of this trip function within a period of eight hours. The Completion Time of 12 hours for Required Action A.2 is based on operating experience in resetting all channels of a protective function and on the low probability of the occurrence of a transient, that could result in steam generator overpressure during this period.

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status in the associated Completion Time, or if one or more steam generators have less than five MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This Surveillance Requirement (SR) verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoints in accordance with the Inservice Testing Program. Reference 2, Section XI, Article IWV-3500, requires that safety and relief valve tests be performed in accordance

Ratings for the No. 1A DG satisfies the requirements of Reference 3 and ratings for the Nos. 1B, 2A, and 2B DGs satisfy the requirements of Reference 4. The continuous service rating for the No. 1A DG is 5400 kW and for the Nos. 1B, 2A, and 2B DGs are 3000 kW.

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in Reference 1, Chapters 6 and 14, assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Sections 3.2, 3.4, and 3.6.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least one train of the onsite or offsite AC sources OPERABLE, during accident conditions in the event of:

a. An assumed loss of all offsite power; and

b. A single failure.

The AC sources satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Electrical Power Distribution System and separate and independent DGs for each train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA.

Qualified offsite circuits are those that are described in the Updated Final Safety Analysis Report (UFSAR) and are part of the licensing basis for the unit.

Each offsite circuit must be capable of maintaining rated frequency and voltage and accepting required loads during an accident, while connected to the ESF buses. Loads are immediately connected to the ESF buses when the buses are

powered from the 500 kV offsite circuits and, when powered from the 69/13.8 kV SMECO offsite circuit after being manually connected, the loads are sequenced onto the ESF bus utilizing the same sequencer used to sequence the loads onto the DG. The SMECO offsite circuit will not be used to carry loads for an operating unit.

The Limiting Condition for Operation (LCO) requires operability of two out of three qualified circuits between the transmission network and the onsite Class 1E AC Electrical Power Distribution System circuits. These circuits consist of two 500 kV circuits via 500 kV/14 kV and 13.8 kV/4.16 kV transformers and the 69 kV SMECO dedicated source (described in Reference 5) via 69 kV/13.8 kV and 13.8 kV/4.16 kV transformers. In addition, each offsite circuit includes one of the two breakers to one 4.16 kV ESF bus. The required circuit breaker to each 4.16 kV ESF bus must be from different 13.8/4.16 unit service transformers for the two required offsite circuits. Thus, each unit is able to align one 4.16 kV bus to one required offsite circuit, and the other 4.16 kV bus to the other required offsite circuit.

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This will be accomplished within 10 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby with the engine at ambient conditions. Additional DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to reject a load ≥ 500 hp without tripping.

Proper sequencing of loads, including shedding of non-essential loads, is a required function for DG OPERABILITY in MODEs 1, 2, and 3.

BASES

The qualified offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ESF bus(es). Qualified offsite circuits are those that are described in the UFSAR and are part of the licensing basis for the unit.

The DG must be capable of starting, accelerating to rated speed and voltage, connecting to its respective ESF bus, and accepting required loads. The DG must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby at ambient conditions.

It is acceptable for trains to be cross-tied during shutdown conditions, allowing a single offsite power circuit to supply all required trains.

The CREVS and CRETS are shared systems with one train of each system connected to an onsite Class 1E AC electrical power distribution subsystem from each unit. Limiting Condition for Operation 3.8.2.c requires one qualified circuit between the offsite transmission network and the other unit's onsite Class 1E AC electrical power distribution subsystems needed to supply power to the CREVS and CRETS to be OPERABLE. Limiting Condition for Operation 3.8.2.d requires one DG from the other unit capable of supplying power to the required CREVS and CRETS to be OPERABLE, if the DG required by LCO 3.8.2.b is not capable of supplying power to the required CREVS and CRETS. These requirements, in conjunction with the requirements for the unit AC electrical power sources in LCO 3.8.2.a and LCO 3.8.2.b, ensure that offsite power is available to both trains and onsite power is available to one train of the CREVS and CRETS, when they are required to be OPERABLE by their respective LCOs (LCOs 3.7.8 and 3.7.9).

BASES

APPLICABILITY

The AC sources required to be OPERABLE in MODEs 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODEs 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

Limiting Condition for Operation 3.0.3 is not applicable while in MODEs 5 or 6. However, since irradiated fuel assembly movement can occur in MODEs 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODEs 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODEs 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

The ACTIONS have been modified by a second Note stating that performance of Required Actions shall not preclude completion of actions to establish a safe conservative position. This clarification is provided to avoid stopping movement of irradiated fuel assemblies while in a non-conservative position based on compliance with the Required Actions.

<u>A.1</u>

An offsite circuit would be considered inoperable, if it was unavailable to one required ESF train. Although two trains may be required by LCO 3.8.10, the remaining train with

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS) and refueling pool during refueling, ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes that have direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the Core Operating Limits Report (COLR). Unit procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control element assemblies and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures. The negative worth of the CEAs may be credited when determining the refueling boron concentrations during full core onloads/offloads only. Unit procedures maintain the number and position of credited CEAs during fuel handling operations.

Reference 1, Appendix 1C, Criterion 27, requires that two independent reactivity control systems of different design principles be provided. One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling pool. The refueling pool is then flooded with borated water from the refueling water tank into the open reactor vessel by gravity feeding or by the use of the Shutdown Cooling (SDC) System pumps.

BASES

The pumping action of the SDC System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and the refueling pool mix the added concentrated boric acid with the water in the RCS. The SDC System is in operation during refueling [see Limiting Condition of Operation (LCO) 3.9.4 and LCO 3.9.5], to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling pool above the COLR limit.

The COLR includes a MODE 6 temperature limitation of ≤ 140°F. This restriction ensures assumptions made for calculating boron concentration and assumptions made in the boron dilution analysis for MODE 6 are preserved.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the unit refueling procedures that demonstrate the correct fuel loading plan (including full core mapping) ensure the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer tube, the refueling pool, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Reference 1, Chapter 14). A detailed discussion of this event is provided in B 3.1.1.

The RCS boron concentration satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

BASES

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS and refueling pool while in MODE 6. The boron concentration limit specified in the COLR ensures a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1 ensures that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS or the refueling pool is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time

BASES

depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This Surveillance Requirement (SR) ensures the coolant boron concentration in the RCS and the refueling pool is within the COLR limits. The coolant boron concentration in each volume is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is therefore a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. Updated Final Safety Analysis Report (UFSAR)

PAGE REPLACEMENT INSTRUCTIONS
Calvert Cliffs Nuclear Power Plant

Technical Specification Bases - Revision 11

Remove and Discard	**Insert**
List of Effective Pages	
LEP-1 through LEP-5	LEP-1 through LEP-5
List of Revisions	
LOR-1	LOR-1
Technical Specification Bases Pages	
B 3.1.7-5	B 3.1.7-5
B 3.2.1-1 through B.3.2.1-6	B 3.2.1-1 through B.3.2.1-7
B 3.2.5-1 through B 3.2.5-5	B 3.2.5-1 through B 3.2.5-6
B 3.3.1-13 through B 3.3.1-34	B 3.3.1-13 through B 3.3.1-34
B 3.3.7-1 and B 3.3.7-2	B 3.3.7-1 and B 3.3.7-2
B 3.3.10-9 and B.3.3.10-10	B 3.3.10-9 and B.3.3.10-10
B 3.7.8-1 through B 3.7.8-7	B 3.7.8-1 through B 3.7.8-7
B 3.7.9-1 through B 3.7.9-3	B 3.7.9-1 through B 3.7.9-4
B 3.9.1-1 and B 3.9.1-2	B 3.9.1-1 and B 3.9.1-2
B 3.9.3-1 through B 3.9.3-6	B 3.9.3-1 through B 3.9.3-7
B 3.9.4-3 and B 3.9.4-4	B 3.9.4-3 and B 3.9.4-4
B 3.9.5-3 and B 3.9.5-4	B 3.9.5-3 and B 3.9.5-4

TECHNICAL SPECIFICATION BASES

LIST OF REVISIONS AND ISSUE DATES

Rev.	Date Issued	Date to NRC
0		May 4, 1998
1	August 28, 1998	October 30, 1998
2	August 28, 1998	October 30, 1998
3	October 28, 1998	October 30, 1998
4	March 16, 1999	October 18, 1999
5	October 18, 1999	October 18, 1999
6	April 14, 2000	October 24, 2000
7	May 18, 2000	October 24, 2000
8	June 29, 2000	October 24, 2000
9	October 24, 2000	October 24, 2000
10	February 1, 2001	
11	March 22, 2001	

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

LEP-1	Rev. 11	B 3.1.2-2	Rev. 2	B 3.2.1-2	Rev. 11
LEP-2	Rev. 11	B 3.1.2-3	Rev. 2	B 3.2.1-3	Rev. 11
LEP-3	Rev. 11	B 3.1.2-4	Rev. 2	B 3.2.1-4	Rev. 11
LEP-4	Rev. 11	B 3.1.2-5	Rev. 2	B 3.2.1-5	Rev. 11
LEP-5	Rev. 11	B 3.1.2-6	Rev. 3	B 3.2.1-6	Rev. 11
i	Rev. 2	B 3.1.3-1	Rev. 2	B 3.2.1-7	Rev. 11
ii	Rev. 2	B 3.1.3-2	Rev. 2	B 3.2.2-1	Rev. 2
iii	Rev. 2	B 3.1.3-3	Rev. 2	B 3.2.2-2	Rev. 8
iv	Rev. 2	B 3.1.3-4	Rev. 2	B 3.2.2-3	Rev. 8
B 2.1.1-1	Rev. 2	B 3.1.3-5	Rev. 2	B 3.2.2-4	Rev. 8
B 2.1.1-2	Rev. 2	B 3.1.4-1	Rev. 2	B 3.2.2-5	Rev. 8
B 2.1.1-3	Rev. 2	B 3.1.4-2	Rev. 2	B 3.2.3-1	Rev. 2
B 2.1.1-4	Rev. 2	B 3.1.4-3	Rev. 2	B 3.2.3-2	Rev. 8
B 2.1.2-1	Rev. 2	B 3.1.4-4	Rev. 2	B 3.2.3-3	Rev. 8
B 2.1.2-2	Rev. 2	B 3.1.4-5	Rev. 2	B 3.2.3-4	Rev. 8
B 2.1.2-3	Rev. 2	B 3.1.4-6	Rev. 2	B 3.2.3-5	Rev. 8
B 2.1.2-4	Rev. 2	B 3.1.4-7	Rev. 2	B 3.2.4-1	Rev. 2
B 3.0-1	Rev. 2	B 3.1.4-8	Rev. 2	B 3.2.4-2	Rev. 8
B 3.0-2	Rev. 2	B 3.1.4-9	Rev. 2	B 3.2.4-3	Rev. 8
B 3.0-3	Rev. 2	B 3.1.4-10	Rev. 2	B 3.2.4-4	Rev. 8
B 3.0-4	Rev. 2	B 3.1.5-1	Rev. 2	B 3.2.4-5	Rev. 8
B 3.0-5	Rev. 2	B 3.1.5-2	Rev. 2	B 3.2.5-1	Rev. 2
B 3.0-6	Rev. 2	B 3.1.5-3	Rev. 2	B 3.2.5-2	Rev. 11
B 3.0-7	Rev. 2	B 3.1.5-4	Rev. 2	B 3.2.5-3	Rev. 11
B 3.0-8	Rev. 2	B 3.1.5-5	Rev. 2	B 3.2.5-4	Rev. 11
B 3.0-9	Rev. 2	B 3.1.6-1	Rev. 2	B 3.2.5-5	Rev. 11
B 3.0-10	Rev. 2	B 3.1.6-2	Rev. 2	B 3.2.5-6	Rev. 11
B 3.0-11	Rev. 2	B 3.1.6-3	Rev. 2	B 3.3.1-1	Rev. 2
B 3.0-12	Rev. 2	B 3.1.6-4	Rev. 2	B 3.3.1-2	Rev. 2
B 3.0-13	Rev. 2	B 3.1.6-5	Rev. 2	B 3.3.1-3	Rev. 2
B 3.0-14	Rev. 2	B 3.1.6-6	Rev. 2	B 3.3.1-4	Rev. 2
B 3.0-15	Rev. 2	B 3.1.6-7	Rev. 2	B 3.3.1-5	Rev. 2
B 3.0-16	Rev. 2	B 3.1.7-1	Rev. 2	B 3.3.1-6	Rev. 2
B 3.0-17	Rev. 2	B 3.1.7-2	Rev. 2	B 3.3.1-7	Rev. 2
B 3.0-18	Rev. 2	B 3.1.7-3	Rev. 5	B 3.3.1-8	Rev. 2
B 3.1.1-1	Rev. 2	B 3.1.7-4	Rev. 2	B 3.3.1-9	Rev. 2
B 3.1.1-2	Rev. 2	B 3.1.7-5	Rev. 11	B 3.3.1-10	Rev. 2
B 3.1.1-3	Rev. 2	B 3.1.8-1	Rev. 2	B 3.3.1-11	Rev. 5
B 3.1.1-4	Rev. 2	B 3.1.8-2	Rev. 2	B 3.3.1-12	Rev. 2
B 3.1.1-5	Rev. 2	B 3.1.8-3	Rev. 2	B 3.3.1-13	Rev. 11
B 3.1.1-6	Rev. 2	B 3.1.8-4	Rev. 2	B 3.3.1-14	Rev. 11
B 3.1.1-7	Rev. 2	B 3.1.8-5	Rev. 2	B 3.3.1-15	Rev. 11
B 3.1.2-1	Rev. 2	B 3.2.1-1	Rev. 2	B 3.3.1-16	Rev. 11

TECHNICAL SPECIFICATIONS BASES
LIST OF EFFECTIVE PAGES

TECHNICAL SPECIFICATIONS BASES
LIST OF EFFECTIVE PAGES

B 3.3.12-3	Rev. 2	B 3.4.10-2	Rev. 2	B 3.4.17-3	Rev. 2
B 3.3.12-4	Rev. 2	B 3.4.10-3	Rev. 2	B 3.5.1-1	Rev. 2
B 3.4.1-1	Rev. 2	B 3.4.10-4	Rev. 2	B 3.5.1-2	Rev. 2
B 3.4.1-2	Rev. 2	B 3.4.11-1	Rev. 2	B 3.5.1-3	Rev. 2
B 3.4.1-3	Rev. 2	B 3.4.11-2	Rev. 2	B 3.5.1-4	Rev. 2
B 3.4.1-4	Rev. 2	B 3.4.11-3	Rev. 2	B 3.5.1-5	Rev. 2
B 3.4.2-1	Rev. 2	B 3.4.11-4	Rev. 2	B 3.5.1-6	Rev. 2
B 3.4.2-2	Rev. 2	B 3.4.11-5	Rev. 2	B 3.5.1-7	Rev. 2
B 3.4.3-1	Rev. 2	B 3.4.11-6	Rev. 2	B 3.5.1-8	Rev. 10
B 3.4.3-2	Rev. 2	B 3.4.11-7	Rev. 2	B 3.5.1-9	Rev. 10
B 3.4.3-3	Rev. 2	B 3.4.12-1	Rev. 2	B 3.5.2-1	Rev. 2
B 3.4.3-4	Rev. 2	B 3.4.12-2	Rev. 2	B 3.5.2-2	Rev. 2
B 3.4.3-5	Rev. 2	B 3.4.12-3	Rev. 2	B 3.5.2-3	Rev. 2
B 3.4.3-6	Rev. 2	B 3.4.12-4	Rev. 2	B 3.5.2-4	Rev. 2
B 3.4.3-7	Rev. 2	B 3.4.12-5	Rev. 6	B 3.5.2-5	Rev. 2
B 3.4.3-8	Rev. 2	B 3.4.12-6	Rev. 2	B 3.5.2-6	Rev. 2
B 3.4.4-1	Rev. 2	B 3.4.12-7	Rev. 2	B 3.5.2-7	Rev. 3
B 3.4.4-2	Rev. 2	B 3.4.12-8	Rev. 2	B 3.5.2-8	Rev. 2
B 3.4.4-3	Rev. 2	B 3.4.12-9	Rev. 2	B 3.5.2-9	Rev. 2
B 3.4.5-1	Rev. 2	B 3.4.12-10	Rev. 2	B 3.5.2-10	Rev. 2
B 3.4.5-2	Rev. 8	B 3.4.12-11	Rev. 2	B 3.5.3-1	Rev. 2
B 3.4.5-3	Rev. 8	B 3.4.12-12	Rev. 2	B 3.5.3-2	Rev. 2
B 3.4.5-4	Rev. 8	B 3.4.12-13	Rev. 2	B 3.5.3-3	Rev. 2
B 3.4.6-1	Rev. 2	B 3.4.13-1	Rev. 2	B 3.5.4-1	Rev. 2
B 3.4.6-2	Rev. 8	B 3.4.13-2	Rev. 10	B 3.5.4-2	Rev. 2
B 3.4.6-3	Rev. 8	B 3.4.13-3	Rev. 2	B 3.5.4-3	Rev. 2
B 3.4.6-4	Rev. 8	B 3.4.13-4	Rev. 2	B 3.5.4-4	Rev. 2
B 3.4.6-5	Rev. 8	B 3.4.13-5	Rev. 5	B 3.5.4-5	Rev. 2
B 3.4.7-1	Rev. 2	B 3.4.14-1	Rev. 2	B 3.5.4-6	Rev. 2
B 3.4.7-2	Rev. 2	B 3.4.14-2	Rev. 2	B 3.5.5-1	Rev. 2
B 3.4.7-3	Rev. 8	B 3.4.14-3	Rev. 2	B 3.5.5-2	Rev. 2
B 3.4.7-4	Rev. 8	B 3.4.14-4	Rev. 2	B 3.5.5-3	Rev. 2
B 3.4.7-5	Rev. 8	B 3.4.14-5	Rev. 2	B 3.5.5-4	Rev. 2
B 3.4.7-6	Rev. 8	B 3.4.15-1	Rev. 2	B 3.5.5-5	Rev. 2
B 3.4.8-1	Rev. 2	B 3.4.15-2	Rev. 2	B 3.6.1-1	Rev. 2
B 3.4.8-2	Rev. 2	B 3.4.15-3	Rev. 2	B 3.6.1-2	Rev. 2
B 3.4.8-3	Rev. 2	B 3.4.15-4	Rev. 3	B 3.6.1-3	Rev. 2
B 3.4.9-1	Rev. 2	B 3.4.15-5	Rev. 2	B 3.6.1-4	Rev. 2
B 3.4.9-2	Rev. 2	B 3.4.16-1	Rev. 2	B 3.6.1-5	Rev. 2
B 3.4.9-3	Rev. 2	B 3.4.16-2	Rev. 2	B 3.6.2-1	Rev. 2
B 3.4.9-4	Rev. 2	B 3.4.16-3	Rev. 2	B 3.6.2-2	Rev. 2
B 3.4.9-5	Rev. 2	B 3.4.17-1	Rev. 2	B 3.6.2-3	Rev. 2
B 3.4.10-1	Rev. 2	B 3.4.17-2	Rev. 2	B 3.6.2-4	Rev. 2

TECHNICAL SPECIFICATIONS BASES
LIST OF EFFECTIVE PAGES

B 3.6.2-5	Rev. 2	B 3.7.1-5	Rev. 9	B 3.7.9-3	Rev. 11
B 3.6.2-6	Rev. 2	B 3.7.2-1	Rev. 2	B 3.7.9-4	Rev. 11
B 3.6.2-7	Rev. 2	B 3.7.2-2	Rev. 2	B 3.7.10-1	Rev. 9
B 3.6.2-8	Rev. 2	B 3.7.2-3	Rev. 2	B 3.7.10-2	Rev. 2
B 3.6.3-1	Rev. 2	B 3.7.2-4	Rev. 2	B 3.7.10-3	Rev. 2
B 3.6.3-2	Rev. 2	B 3.7.2-5	Rev. 2	B 3.7.11-1	Rev. 2
B 3.6.3-3	Rev. 2	B 3.7.3-1	Rev. 2	B 3.7.11-2	Rev. 2
B 3.6.3-4	Rev. 2	B 3.7.3-2	Rev. 2	B 3.7.11-3	Rev. 2
B 3.6.3-5	Rev. 2	B 3.7.3-3	Rev. 2	B 3.7.11-4	Rev. 2
B 3.6.3-6	Rev. 2	B 3.7.3-4	Rev. 2	B 3.7.12-1	Rev. 2
B 3.6.3-7	Rev. 2	B 3.7.3-5	Rev. 2	B 3.7.12-2	Rev. 2
B 3.6.3-8	Rev. 2	B 3.7.3-6	Rev. 2	B 3.7.12-3	Rev. 2
B 3.6.3-9	Rev. 2	B 3.7.3-7	Rev. 2	B 3.7.12-4	Rev. 2
B 3.6.3-10	Rev. 2	B 3.7.3-8	Rev. 2	B 3.7.13-1	Rev. 8
B 3.6.4-1	Rev. 2	B 3.7.3-9	Rev. 2	B 3.7.13-2	Rev. 8
B 3.6.4-2	Rev. 2	B 3.7.3-10	Rev. 2	B 3.7.13-3	Rev. 8
B 3.6.4-3	Rev. 2	B 3.7.4-1	Rev. 2	B 3.7.14-1	Rev. 2
B 3.6.5-1	Rev. 2	B 3.7.4-2	Rev. 8	B 3.7.14-2	Rev. 2
B 3.6.5-2	Rev. 2	B 3.7.4-3	Rev. 2	B 3.7.14-3	Rev. 2
B 3.6.5-3	Rev. 3	B 3.7.4-4	Rev. 2	B 3.7.15-1	Rev. 2
B 3.6.6-1	Rev. 2	B 3.7.5-1	Rev. 2	B 3.7.15-2	Rev. 2
B 3.6.6-2	Rev. 2	B 3.7.5-2	Rev. 2	B 3.7.15-3	Rev. 2
B 3.6.6-3	Rev. 2	B 3.7.5-3	Rev. 2	B 3.7.15-4	Rev. 2
B 3.6.6-4	Rev. 5	B 3.7.5-4	Rev. 2	B 3.8.1-1	Rev. 5
B 3.6.6-5	Rev. 2	B 3.7.5-5	Rev. 2	B 3.8.1-2	Rev. 5
B 3.6.6-6	Rev. 2	B 3.7.6-1	Rev. 5	B 3.8.1-3	Rev. 2
B 3.6.6-7	Rev. 2	B 3.7.6-2	Rev. 2	B 3.8.1-4	Rev. 10
B 3.6.6-8	Rev. 2	B 3.7.6-3	Rev. 5	B 3.8.1-5	Rev. 7
B 3.6.6-9	Rev. 2	B 3.7.6-4	Rev. 5	B 3.8.1-6	Rev. 7
B 3.6.7-1	Rev. 2	B 3.7.6-5	Rev. 5	B 3.8.1-7	Rev. 3
B 3.6.7-2	Rev. 2	B 3.7.7-1	Rev. 5	B 3.8.1-8	Rev. 3
B 3.6.7-3	Rev. 2	B 3.7.7-2	Rev. 5	B 3.8.1-9	Rev. 3
B 3.6.7-4	Rev. 2	B 3.7.7-3	Rev. 2	B 3.8.1-10	Rev. 3
B 3.6.7-5	Rev. 2	B 3.7.7-4	Rev. 2	B 3.8.1-11	Rev. 3
B 3.6.7-6	Rev. 2	B 3.7.8-1	Rev. 8	B 3.8.1-12	Rev. 3
B 3.6.8-1	Rev. 2	B 3.7.8-2	Rev. 11	B 3.8.1-13	Rev. 3
B 3.6.8-2	Rev. 2	B 3.7.8-3	Rev. 11	B 3.8.1-14	Rev. 3
B 3.6.8-3	Rev. 2	B 3.7.8-4	Rev. 11	B 3.8.1-15	Rev. 3
B 3.6.8-4	Rev. 2	B 3.7.8-5	Rev. 11	B 3.8.1-16	Rev. 3
B 3.7.1-1	Rev. 2	B 3.7.8-6	Rev. 11	B 3.8.1-17	Rev. 3
B 3.7.1-2	Rev. 9	B 3.7.8-7	Rev. 11	B 3.8.1-18	Rev. 3
B 3.7.1-3	Rev. 9	B 3.7.9-1	Rev. 2	B 3.8.1-19	Rev. 3
B 3.7.1-4	Rev. 10	B 3.7.9-2	Rev. 11	B 3.8.1-20	Rev. 3

TECHNICAL SPECIFICATIONS BASES
LIST OF EFFECTIVE PAGES

B 3.8.1-21	Rev. 3	B 3.8.6-7	Rev. 2	B 3.9.5-3	Rev. 11
B 3.8.1-22	Rev. 3	B 3.8.7-1	Rev. 2	B 3.9.5-4	Rev. 2
B 3.8.1-23	Rev. 3	B 3.8.7-2	Rev. 2	B 3.9.5-5	Rev. 2
B 3.8.1-24	Rev. 3	B 3.8.7-3	Rev. 2	B 3.9.6-1	Rev. 2
B 3.8.1-25	Rev. 3	B 3.8.7-4	Rev. 2	B 3.9.6-2	Rev. 2
B 3.8.1-26	Rev. 5	B 3.8.8-1	Rev. 2	B 3.9.6-3	Rev. 2
B 3.8.1-27	Rev. 5	B 3.8.8-2	Rev. 2		
B 3.8.1-28	Rev. 5	B 3.8.8-3	Rev. 2		
B 3.8.1-29	Rev. 5	B 3.8.9-1	Rev. 5		
B 3.8.2-1	Rev. 2	B 3.8.9-2	Rev. 2		
B 3.8.2-2	Rev. 2	B 3.8.9-3	Rev. 2		
B 3.8.2-3	Rev. 10	B 3.8.9-4	Rev. 2		
B 3.8.2-4	Rev. 5	B 3.8.9-5	Rev. 2		
B 3.8.2-5	Rev. 5	B 3.8.9-6	Rev. 2		
B 3.8.2-6	Rev. 5	B 3.8.9-7	Rev. 2		
B 3.8.3-1	Rev. 2	B 3.8.9-8	Rev. 2		
B 3.8.3-2	Rev. 2	B 3.8.9-9	Rev. 2		
B 3.8.3-3	Rev. 2	B 3.8.9-10	Rev. 2		
B 3.8.3-4	Rev. 2	B 3.8.10-1	Rev. 5		
B 3.8.3-5	Rev. 2	B 3.8.10-2	Rev. 5		
B 3.8.3-6	Rev. 2	B 3.8.10-3	Rev. 5		
B 3.8.3-7	Rev. 2	B 3.8.10-4	Rev. 5		
B 3.8.3-8	Rev. 3	B 3.8.10-5	Rev. 5		
B 3.8.3-9	Rev. 2	B 3.9.1-1	Rev. 11		
B 3.8.4-1	Rev. 2	B 3.9.1-2	Rev. 10		
B 3.8.4-2	Rev. 2	B 3.9.1-3	Rev. 10		
B 3.8.4-3	Rev. 2	B 3.9.1-4	Rev. 10		
B 3.8.4-4	Rev. 2	B 3.9.2-1	Rev. 2		
B 3.8.4-5	Rev. 2	B 3.9.2-2	Rev. 2		
B 3.8.4-6	Rev. 2	B 3.9.2-3	Rev. 2		
B 3.8.4-7	Rev. 2	B 3.9.3-1	Rev. 11		
B 3.8.4-8	Rev. 2	B 3.9.3-2	Rev. 11		
B 3.8.4-9	Rev. 2	B 3.9.3-3	Rev. 11		
B 3.8.5-1	Rev. 2	B 3.9.3-4	Rev. 11		
B 3.8.5-2	Rev. 2	B 3.9.3-5	Rev. 11		
B 3.8.5-3	Rev. 2	B 3.9.3-6	Rev. 11		
B 3.8.5-4	Rev. 2	B 3.9.3-7	Rev. 11		
B 3.8.6-1	Rev. 2	B 3.9.4-1	Rev. 2		
B 3.8.6-2	Rev. 2	B 3.9.4-2	Rev. 2		
B 3.8.6-3	Rev. 2	B 3.9.4-3	Rev. 11		
B 3.8.6-4	Rev. 2	B 3.9.4-4	Rev. 11		
B 3.8.6-5	Rev. 2	B 3.9.5-1	Rev. 2		
B 3.8.6-6	Rev. 2	B 3.9.5-2	Rev. 2		

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification of the position of each partially or fully withdrawn full-length or part-length CEA is necessary to ensure that the minimum negative reactivity requirements for insertion on a trip are preserved. A 2-hour Frequency is sufficient for the operator to verify that each CEA position is within the acceptance criteria.

SR 3.1.7.2

Prior demonstration that each CEA to be withdrawn from the core during PHYSICS TESTS is capable of full insertion, when tripped from at least a 50% withdrawn position, ensures that the CEA will insert on a trip signal. The Frequency ensures that the CEAs are OPERABLE prior to reducing SDM to less than the limits of LCO 3.1.1.

The SR is modified by a Note that allows the SR to not be performed during initial power escalation following a refueling outage if SR 3.1.4.6 has been met during that refueling outage. This allows the CEA drop time test, which also proves the CEAs are trippable, to be credited for this SR.

REFERENCES

1. 10 CFR Part 50
2. Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," August 1978
3. UFSAR

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Linear Heat Rate (LHR)

BASES

BACKGROUND

The purpose of this Limiting Condition for Operation (LCO) is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow accident (LOFA), ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protective System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting less than optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution satisfies this LCO. The limiting safety system settings (LSSS) and this LCO are based on the accident analyses (Reference 1, Chapter 14), so that specified acceptable fuel design limits (SAFDLs) are not exceeded as a result of anticipated operational occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

BASES

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on linear heat rate (LHR) and departure from nucleate boiling (DNB).

The limits on LHR, Total Planar Radial Peaking Factor (F_{xy}^T), Total Integrated Radial Peaking Factor (F_r^T), AZIMUTHAL POWER TILT (T_q), and AXIAL SHAPE INDEX (ASI) represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Below 20% power, ASI limits for the LHR and DNB LCO are not required. At low powers, the axial power distribution (APD) trip will limit the allowed ASI during operation. The core reload analysis verifies that ASI limits for the LHR and DNB LCOs are not necessary below 20% power.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LHR is within its limits. The Excore Detector Monitoring System performs this function by continuously monitoring ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the Core Operating Limit Report (COLR).

In conjunction with the use of the Excore Detector Monitoring System and in establishing ASI limits, the following assumptions are made:

a. The CEA insertion limits of LCOs 3.1.5 and 3.1.6 are satisfied;

b. The T_q restrictions of LCO 3.2.4 are satisfied; and

c. F_{xy}^T is within the limits of LCO 3.2.2.

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors and alarms that

have been established for the individual incore detector segments, ensuring that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include allowances described in the COLR.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs (Reference 1, Appendix 1C, Criterion 6). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Reference 2);

b. During a LOFA, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected CEA accident, the energy input to the fuel must not exceed the accepted limits (Reference 1, Section 14.13); and

d. The control rods must be capable of shutting down the reactor with a minimum required SHUTDOWN MARGIN (SDM) with the highest worth control rod stuck fully withdrawn (Reference 1, Appendix 1C, Criterion 29).

The power density at any point in the core must be limited to maintain the fuel design criteria (Reference 2). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by accident analyses (Reference 1, Chapter 14), with due regard for the correlations between measured quantities, the power distribution, and uncertainties in determining the power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Reference 2). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy-water reaction.

BASES

The LCOs governing LHR, ASI, and the Reactor Coolant System (RCS) ensure that these criteria are met as long as the core is operated within the ASI, F^T_{xy}, F^T_r, and T_q limits specified in the COLR. The latter are process variables that characterize the three-dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

Below 20% power, ASI limits for the LHR and DNB LCO are not required. At low powers, the APD trip will limit the allowed ASI during operation. The core reload analysis verifies that ASI limits for the LHR and DNB LCOs are not necessary below 20% power.

Fuel cladding damage does not normally occur while the unit is operating at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, if an accident or AOO occurs from initial conditions outside the limits of these LCOs. The potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and can correspondingly increase local LHR.

The LHR satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNB ratio operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limitation on the LHR ensures that, in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F. However, fuel cladding damage does not normally occur when outside the LCO limit if an accident does not occur.

APPLICABILITY

In MODE 1, power distribution must be maintained within the limits assumed in the accident analysis to ensure that fuel damage does not result following an AOO. In other MODEs, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution.

ACTIONS

A.1

With the LHR exceeding its limit, excessive fuel damage could occur following an accident. In this Condition, prompt action must be taken to restore the LHR to within the specified limits. One hour to restore the LHR to within its specified limits is reasonable and ensures that the core does not continue to operate in this Condition. The 1-hour Completion Time also allows the operator sufficient time for evaluating core conditions and for initiating proper corrective actions.

B.1

If the LHR cannot be returned to within its specified limits, THERMAL POWER must be reduced. Since ASI limits for LHR are not required below 20% Rated Thermal Power (RTP), then the actions of A.1 can be met by reducing power to < 20% RTP. Reducing THERMAL POWER to < 20% RTP provides reasonable assurance that the core is operating farther from thermal limits and places the core in a conservative condition. This action is also consistent with the required actions for the SAFDL on DNB. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach the applicable power level from full power MODE 1 conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

A Note was added to the Surveillance Requirements (SRs) to require LHR to be determined by either the Excore Detector Monitoring System or the Incore Detector Monitoring System.

SR 3.2.1.1

The periodic SR to verify the value of F_{xy}^{T} ensures that the LHR remains within the range assumed in the analysis. Determining the measured F_{xy}^{T} every 72 hours when the excores are used to monitor LHR ensures the power distribution parameters are within limits when full core mapping is not being used.

Performance of the SR every 72 hours of accumulated operation in MODE 1 provides reasonable assurance that

unacceptable changes in the F_{xy}^{T} and LHR are promptly detected.

The SR is modified by a Note that only requires the SR to be performed when the excores are being used to determine LHR. This SR is not required when the LHR is being measured by the incores, which is a more accurate measure of Core Power Distributions.

SR 3.2.1.2

This SR requires verification that the ASI alarm setpoints are within the limits specified in the COLR. Performance of this SR ensures that the Excore Detector Monitoring System can accurately monitor the LHR, and provide alarms when LHR is not within limits. Therefore, this SR is only applicable when the Excore Detector Monitoring System is being used to determine the LHR. The F_{xy}^{T} value determined by SR 3.2.1.1 is used in the derivation of the ASI alarm setpoint specified in the COLR. The 31-day Frequency is appropriate for this SR because it is consistent with the requirements of SR 3.3.1.3 for calibration of the excore detectors using the incore detectors.

The SR is modified by a Note that states that the SR is only applicable when the Excore Detection Monitoring System is being used to determine LHR. The reason for the Note is that the excore detectors input neutron flux information into the ASI calculation.

SR 3.2.1.3 and SR 3.2.1.4

Continuous monitoring of the LHR is provided by the Incore Detector Monitoring System and the Excore Detector Monitoring System. Either of these two core power distribution monitoring systems provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its specified limits.

Performance of these SRs verifies that the Incore Detector Monitoring System can accurately monitor LHR. Therefore, they are only applicable when the Incore Detector Monitoring System is being used to determine the LHR.

BASES

A 31-day Frequency is consistent with the historical testing frequency of the incore detector monitoring system. The SRs are modified by two Notes. Note 1 allows the SRs to be performed only when the Incore Detector Monitoring System is being used to determine LHR. Note 2 states that the SRs are not required to be performed when THERMAL POWER is < 20% RTP. The accuracy of the neutron flux information from the incore detectors is not reliable at THERMAL POWER < 20% RTP.

REFERENCES

1. Updated Final Safety Analysis Report (UFSAR)
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 AXIAL SHAPE INDEX (ASI)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analysis. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, LOFA, ejected control element assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protective System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The LSSS and this LCO are based on the accident analyses (Reference 1, Chapter 14), so that SAFDLs are not exceeded as a result of AOOs, and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on LHR and DNB.

The limits on LHR, F_{xy}^{T}, F_{r}^{T}, T_q, and ASI represent limits within which the LHR algorithms are valid. These limits are obtained directly from the core reload analysis.

Below 20% power, ASI limits for the LHR and DNB LCO are not required. At low powers, the APD trip will limit the allowed ASI during operation. The core reload analysis verifies that ASI limits for the LHR and DNB LCOs are not necessary below 20% power.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the LHR does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the ASI with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the ASI is maintained within the allowable limits specified in the COLR.

In conjunction with the use of the Excore Detector Monitoring System and in establishing the ASI limits, the following conditions are assumed:

a. The CEA insertion limits of LCOs 3.1.5 and 3.1.6 are satisfied;

b. The T_q restrictions of LCO 3.2.4 are satisfied; and

c. F_{xy}^{T} does not exceed the limits of LCO 3.2.2.

The Incore Detector Monitoring System continuously provides a more direct measure of the peaking factors, and the alarms that have been established for the individual incore detector segments ensure that the peak LHR is maintained within the limits specified in the COLR. The setpoints for these alarms include allowances described in the COLR.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Reference 1, Appendix 1C, Criterion 6). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from

reaching levels that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Reference 2);

b. During a LOFA, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected CEA accident, the energy input to the fuel must not exceed the acceptable limits (Reference 1, Section 14.13); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Reference 1, Appendix 1C, Criterion 29).

The power density at any point in the core must be limited to maintain the fuel design criteria (Reference 2). This limitation is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Reference 1, Chapter 14), with due regard for the correlations among measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum LHGR so that the peak cladding temperature does not exceed 2200°F (Reference 2). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy-water reaction.

The LCOs governing LHR, ASI, and the RCS ensure that these criteria are met as long as the core is operated within the ASI, F_{xy}^T, and F_r^T limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three-dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

BASES

Below 20% power, ASI limits for the LHR and DNB LCO are not required. At low powers, the APD trip will limit the allowed ASI during operation. The core reload analysis verifies that ASI limits for the LHR and DNB LCOs are not necessary below 20% power.

Fuel cladding damage does not normally occur while the reactor is operating at conditions outside these LCOs during normal operation. Fuel cladding damage results, however, when an accident or AOO occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

The ASI satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNB ratio operating limits. These power distribution LCO limits, except T_q, are provided in the COLR. The limitation on LHR ensures that in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

The limitation on ASI, along with the limitations of LCO 3.3.1, represents a conservative envelope of operating conditions consistent with the assumptions that have been analytically-demonstrated adequate for maintaining an acceptable minimum DNB ratio throughout all AOOs. Of these, the loss of flow transient is the most limiting. Operation of the core with conditions within the specified limits ensures that an acceptable minimum margin from DNB conditions is maintained in the event of any AOO, including a loss of flow transient.

APPLICABILITY

In MODE 1 with THERMAL POWER > 20% RTP, power distribution must be maintained within the limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In other MODEs, this LCO does not apply because THERMAL POWER is not sufficient to require a limit on the core power distribution. Below 20% RTP, the incore detector accuracy is not reliable.

ACTIONS

A.1

Operating the core within ASI limits specified in the COLR and within the limits of LCO 3.3.1 ensures an acceptable margin for DNB and for maintaining local power density in the event of an AOO. Maintaining ASI within limits also ensures that the limits of Reference 2 are not exceeded during accidents. The Required Actions to restore ASI must be completed within 2 hours to limit the duration the plant is operated outside the initial conditions assumed in the accident analyses. In addition, this Completion Time is sufficiently short that the xenon distribution in the core cannot change significantly.

B.1

If the ASI cannot be restored to within its specified limits, or ASI cannot be determined because of Excore Detector Monitoring System inoperability, core power must be reduced. Reducing THERMAL POWER to ≤ 20% RTP provides reasonable assurance that the core is operating farther from thermal limits and places the core in a conservative condition. Four hours is a reasonable amount of time, based on operating experience, to reduce THERMAL POWER to ≤ 20% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

Verifying that the ASI is within the specified limits provides reasonable assurance that the core is not approaching DNB conditions. A Frequency of 12 hours is adequate for the operator to identify trends in conditions that result in an approach to the ASI limits, because the mechanisms that affect the ASI, such as xenon redistribution or CEA drive mechanism malfunctions, cause the ASI to change slowly and should be discovered before the limits are exceeded.

BASES

REFERENCES

1. UFSAR
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

8. APD-High Trip

The APD-High trip ensures that excessive axial peaking, such as that due to axial xenon oscillations, will not cause fuel damage. It ensures that neither a DNBR less than the SL, nor a peak linear heat rate that corresponds to the temperature for fuel centerline melting, will occur. This trip is the primary protection against fuel centerline melting. While no event specifically credits the Axial Flux Offset trip, the ASI limits established by this trip provide ASI limits for safety and setpoint analyses.

9. Thermal Margin

a. TM/LP Trip

The TM/LP trip prevents exceeding the DNBR SL during AOOs and aids the ESFAS during certain accidents. The following events require TM/LP trip protection:

- RCS depressurization (inadvertent safety or power-operated relief valves opening);
- Steam generator tube rupture; and
- LOCA accident.

The first event is an AOOs, and fuel integrity is maintained. The second and third events are accidents, and limited fuel damage may occur, although only the LOCA is expected to result in fuel damage. The trip is initiated whenever the RCS pressure signal drops below a minimum value (P_{min}) or a computed value (P_{var}) as described below, whichever is higher. The setpoint is a Function of Q power, ASI, reactor inlet (cold leg) temperature, and the number of RCPs operating.

The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT (T_q), and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip signal. In addition, CEA group sequencing in accordance with LCO 3.1.7 is assumed. Finally,

the maximum insertion of CEA banks that can occur during any AOO prior to a Power Level-High trip is assumed.

b. ASGT

The ASGT provides protection for those AOOs associated with secondary system malfunctions that result in asymmetric primary coolant temperatures. The most limiting event is closure of a single main steam isolation valve (MSIV). Asymmetric Steam Generator Transient is provided by comparing the secondary pressure in both steam generators in the TM/LP trip calculator. If the pressure in either exceeds that in the other by the trip setpoint, a TM/LP trip will result.

10. Loss of Load

The Loss of Load trip causes a trip when operating above 15% of RTP. This trip provides turbine protection, reduces the severity of the ensuing transient, and helps avoid the lifting of the main steam safety valves during the ensuing transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability is required to enhance overall plant equipment service life and reliability.

Operating Bypasses

The operating bypasses are addressed in footnotes to Table 3.3.1-1. They are not otherwise addressed as specific table entries.

The automatic bypass removal features must function as a backup to manual actions for all trips credited in safety analyses to ensure the trip Functions are not operationally bypassed when the safety analysis assumes the Functions are not bypassed. The RPS operating bypasses are:

Zero power mode bypass (ZPMB) removal of the TM/LP, ASGT, and reactor coolant low flow trips when NUCLEAR INSTRUMENT POWER is < 1E-4% RTP. This bypass is manually enabled below

the specified setpoint to permit low power testing. The wide range NI Level 1 bistable in the wide range drawer provides a signal to auxiliary logic, which then permits manual bypassing below the setpoint and removes the bypass above the setpoint.

Power rate of change bypass removal — The Rate of Change of Power-High trip is automatically bypassed at < 1E-4% RTP, as sensed by the wide range NI Level 1 bistable, and at > 12% RTP by the linear range NI Level 1 bistable, mounted in their respective NI drawers (Reference 5). Automatic bypass removal is also effected by these bistables when conditions are no longer satisfied.

Loss of Load and APD-High trip bypass removal — The Loss of Load and APD-High trips are automatically bypassed when at < 15% RTP as sensed by the linear range NI Level 1 bistable. The bypass is automatically removed by this bistable above the setpoint. This same bistable is used to bypass the Rate of Change of Power-High trip.

Steam Generator Pressure-Low trip bypass removal. The Steam Generator Pressure-Low trip is manually enabled below the pretrip setpoint. The permissive signal is removed, and the bypass automatically removed, when the Steam Generator Pressure-Low trip is above the pretrip setpoint.

The RPS instrumentation satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any required portion of the instrument channel renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. The specific criteria for determining channel OPERABILITY differ slightly between Functions. These criteria are discussed on a Function-by-Function basis below.

Actions allow trip channel bypass of individual instrument channels, but administrative controls prevent operation with a second channel in the same Function bypassed. Plants are restricted to 48 hours in a trip bypass condition before

either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic).

Only the Allowable Values are specified for each RPS trip Function in the LCO. Nominal trip setpoints are established for the Functions via the plant-specific procedures. The nominal setpoints are selected to ensure the plant parameters do not exceed the Allowable Value if the bistable trip unit is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the plant-specific setpoint calculations. Each nominal trip setpoint is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument channel uncertainties appropriate to the trip Function. These uncertainties are defined in Reference 4. The nominal trip setpoint entered into a bistable is more conservative than that specified by the Allowable Value. A channel is inoperable if its actual setpoint is not within its required Allowable Value.

The following Bases for each trip Function identify the above RPS trip Function criteria items that are applicable to establish the trip Function OPERABILITY.

1. <u>Power Level-High Trip</u>

   This LCO requires all four instrument channels of the Power Level-High trip to be OPERABLE in MODEs 1 and 2.

   The Allowable Value is high enough to provide an operating envelope that prevents unnecessary Power Level-High trips during normal plant operations. The Allowable Value is low enough for the system to maintain a margin to unacceptable fuel cladding damage should a CEA ejection accident occur.

   The Power Level-High trip setpoint is operator adjustable and can be set at a fixed increment above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL

POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases. The trip setpoint has a maximum and a minimum setpoint.

Adding to this maximum value the possible variation in trip setpoint due to calibration and instrument errors, the maximum actual steady state THERMAL POWER level at which a trip would be actuated is 109% RTP, which is the value used in the safety analyses.

To account for these errors, the safety analysis minimum value is 40% RTP. The 10% step increase in trip setpoint is a maximum value assumed in the safety analysis. There is no uncertainty applied to the step in the safety analyses.

2. Rate of Change of Power-High Trip

This LCO requires four instrument channels of Rate of Change of Power-High trip to be OPERABLE in MODEs 1 and 2.

The high power rate of change trip serves as a backup to the administratively-enforced startup rate limit. The Function is not credited in the accident analyses; therefore, the Allowable Value for the trip is not derived from analytical limits.

3. Reactor Coolant Flow-Low Trip

This LCO requires four instrument channels of Reactor Coolant Flow-Low trip to be OPERABLE in MODEs 1 and 2.

The trip may be manually bypassed when NUCLEAR INSTRUMENT POWER falls below 1E-4% RTP. This operating bypass is part of the ZPMB circuitry, which also bypasses the TM/LP trip and provides a ΔT power block signal to the Q power select logic. The ZPMB allows low power physics testing at reduced RCS temperatures and pressures. It also allows heatup and cooldown with shutdown CEAs withdrawn.

This trip is set high enough to maintain fuel integrity during a loss of flow condition. The setting is low enough to allow for normal operating fluctuations from offsite power. To account for analysis uncertainty, the value in the safety analysis is 93% of design flow. Reactor Coolant System flow is maintained above design flow by LCO 3.4.1.

4. <u>Pressurizer Pressure-High Trip</u>

This LCO requires four instrument channels of Pressurizer Pressure-High trip to be OPERABLE in MODEs 1 and 2.

The Allowable Value is set high enough to allow for pressure increases in the RCS during normal operation (i.e., plant transients) not indicative of an abnormal condition. The setting is below the lift setpoint of the pressurizer safety valves and low enough to initiate a reactor trip when an abnormal condition is indicated. The analysis setpoint includes allowance for harsh environment, where appropriate.

The Pressurizer Pressure-High trip concurrent with power-operated relief valve operation avoids unnecessary operation of the pressurizer safety valves (Reference 5).

5. <u>Containment Pressure-High Trip</u>

This LCO requires four instrument channels of Containment Pressure-High trip to be OPERABLE in MODEs 1 and 2.

The Allowable Value is high enough to allow for small pressure increases in Containment, expected during normal operation (i.e., plant heatup) that are not indicative of an abnormal condition. The setting is low enough to initiate a reactor trip to prevent containment pressure from exceeding design pressure following a DBA.

6. Steam Generator Pressure-Low Trip

This LCO requires four instrument channels of Steam Generator Pressure-Low trip per steam generator to be OPERABLE in MODEs 1 and 2.

The Allowable Value is sufficiently below the full load operating value for steam pressure so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of excessive steam demand. Since excessive steam demand causes the RCS to cool down, resulting in positive reactivity addition to the core in the presence of a negative moderator temperature coefficient, a reactor trip is required to offset that effect.

The analysis setpoint value includes harsh environment uncertainties, where appropriate.

The Function may be manually bypassed as steam generator pressure is reduced during controlled plant shutdowns. This operating bypass is permitted at a preset steam generator pressure. The bypass, in conjunction with the ZPMB, allows testing at low temperatures and pressures, and heatup and cooldown with the shutdown CEAs withdrawn. From a bypass condition, the trip will be automatically reinstated as steam generator pressure increases above the preset pressure.

7. Steam Generator Level-Low Trip

This LCO requires four instrument channels of Steam Generator Level-Low per steam generator to be OPERABLE in MODEs 1 and 2.

The Allowable Value is sufficiently below the normal operating level for the steam generators so as not to cause a reactor trip during normal plant operations. The trip setpoint is high enough to ensure a reactor trip signal is generated to prevent operation with the steam generator water level below the minimum volume required for adequate heat removal capacity, and

ensures that the pressure of the RCS will not exceed its SL. The specified setpoint, in combination with the Auxiliary Feedwater Actuation System (AFAS), ensures that sufficient water inventory exists in both steam generators to remove decay heat following a Loss of Main Feedwater Flow event.

8. APD-High Trip

This LCO requires four instrument channels of APD-High trip to be OPERABLE in MODE 1, NUCLEAR INSTRUMENT POWER ≥ 15% RTP.

The Allowable Value curve was derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies and the uncertainty associated with the excore to incore ASI relationship.

The APD-High trip is automatically bypassed at < 15% RTP, as measured by the NIs, where it is not required for reactor protection (Reference 5).

9. Thermal Margin

a. TM/LP Trip

This LCO requires four instrument channels of TM/LP trip to be OPERABLE in MODEs 1 and 2.

The Allowable Value includes allowances for equipment response time, measurement uncertainties, processing error, and a further allowance to compensate for the time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the SLs.

This trip may be manually bypassed when NUCLEAR INSTRUMENT POWER falls below 1E-4% RTP. This operating bypass is part of the ZPMB circuitry, which also bypasses the Reactor Coolant Flow-Low trip and provides a ΔT power block signal to the Q power select logic (Reference 5). The ZPMB allows low power physics testing at reduced RCS temperatures and pressures. It also allows

heatup and cooldown with shutdown CEAs withdrawn.

b. ASGT

This LCO requires four instrument channels of ASGT to be OPERABLE in MODEs 1 and 2.

The Allowable Value is high enough to avoid trips caused by normal operation and minor transients, but ensures DNBR protection in the event of DBAs. The difference between the Allowable Value and the analysis setpoint allows for instrument uncertainty.

The trip may be manually bypassed when NUCLEAR INSTRUMENT POWER falls below 1E-4% RTP as part of the ZPMB circuitry operating bypass. The Steam Generator Pressure Difference is subject to the ZPMB, since it is an input to the TM/LP trip and is not required for protection at low power levels (Reference 5).

10. Loss of Load

The LCO requires four Loss of Load instrument channels to be OPERABLE in MODE 1, NUCLEAR INSTRUMENT POWER ≥ 15% RTP.

The Loss of Load trip is automatically bypassed when NUCLEAR INSTRUMENT POWER falls below 15%, as measured by NIs, to allow loading the turbine.

Bypasses

The LCO on automatic bypass removal features requires that the automatic bypass removal feature of all four operating bypass channels be OPERABLE for each RPS Function with an operating bypass in the MODEs addressed in the specific LCO for each Function. All four automatic bypass removal features must be OPERABLE to ensure that none of the four RPS instrument channels are inadvertently bypassed.

BASES

The LCO applies to the automatic bypass removal feature only. If the bypass channel is failed so as to prevent entering a bypass condition, operation may continue.

APPLICABILITY

This LCO is applicable in accordance with Table 3.3.3-1. Most RPS trip functions are required to be OPERABLE in MODEs 1 and 2 because the reactor is critical in these MODEs. The trips are designed to take the reactor subcritical, maintaining the SLs during AOOs and assisting the ESFAS in providing acceptable consequences during accidents. Exceptions are addressed in footnotes to the table. Exceptions to this APPLICABILITY are:

- The APD-High and Loss-of-Load trips are only applicable in MODE 1, NUCLEAR INSTRUMENT POWER ≥ 15% RTP because they are automatically bypassed at < 15% RTP, as measured by NIs, where they are no longer needed.
- The Rate of Change of Power-High trip, RPS logic, RTCBs, and manual trip are also required in MODEs 3, 4, and 5, with the RTCBs closed, to provide protection for boron dilution and CEA withdrawal events. The Rate of Change of Power-High trip in these lower MODEs is addressed in LCO 3.3.2. The RPS logic in MODEs 1, 2, 3, 4, and 5 is addressed in LCO 3.3.3.

Most trip functions are not required to be OPERABLE in MODEs 3, 4, and 5. In MODEs 3, 4, and 5, the emphasis is placed on return to power events. The reactor is protected in these MODEs by ensuring adequate SHUTDOWN MARGIN (SDM).

ACTIONS

The most common causes of instrument channel inoperability are outright failure or drift of the bistable trip unit or measurement channel sufficient to exceed the tolerance allowed by Reference 4. Typically, the drift is found to be small which, at worst, results in a delay of actuation rather than a total loss of Function. This determination is generally made during the performance of a CHANNEL CALIBRATION when the process instrument is set up for adjustment to bring it to within specification. Sensor Drift could also be identified during the CHANNEL CHECKS. CHANNEL FUNCTIONAL TESTs identify bistable trip unit drift. If the trip setpoint is less conservative than the Allowable Value in Table 3.3.1-1, the instrument channel is declared

inoperable immediately, and the appropriate Condition(s) must be entered immediately.

In the event that either an instrument channel's trip setpoint is found nonconservative with respect to the Allowable Value or the transmitter, instrument loop, signal processing electronics, RPS bistable trip unit, or applicable automatic bypass removal feature when bypass is in effect, is found inoperable, then all affected Functions provided by that channel must be declared inoperable, and the plant must enter the Condition for the particular protection Function affected.

When the number of inoperable instrument channels in a trip Function exceeds that specified in any related Condition associated with the same trip Function, the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered, if applicable, in the current MODE of operation.

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Times of each inoperable Function will be tracked separately for each Function, starting from the time the Condition was entered.

A.1, A.2.1, and A.2.2

Condition A applies to the failure of a single instrument channel in any RPS automatic trip Function. Reactor Protective System coincidence logic is normally two-out-of-four.

If one RPS bistable trip unit or associated measurement channel is inoperable, startup or power operation is allowed to continue, providing the inoperable bistable trip unit is placed in bypass or trip within 1 hour (Required Action A.1).

The Completion Time of 1 hour allotted to restore, bypass, or trip the instrument channel is sufficient to allow the operator to take all appropriate actions for the failed

channel, while ensuring that the risk involved in operating with the failed channel is acceptable.

The failed instrument channel is restored to OPERABLE status or is placed in trip within 48 hours (Required Action A.2.1 or Required Action A.2.2). Required Action A.2.1 restores the full capability of the Function.

Required Action A.2.2 places the Function in a one-out-of-three configuration. In this configuration, common cause failure of dependent channels cannot prevent a trip.

The Completion Time of 48 hours is based on operating experience, which has demonstrated that a random failure of a second instrument channel occurring during the 48-hour period is a low probability event.

B.1 and B.2

Condition B applies to the failure of two instrument channels in any RPS automatic trip Function.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODEs, even though two instrument channels are inoperable, with one channel bypassed and one tripped. MODE changes in this configuration are allowed to permit maintenance and testing on one of the inoperable channels. In this configuration, the protective system is in a one-out-of-two logic, and the probability of a common cause failure affecting both of the OPERABLE channels during the 48 hours permitted is remote.

Required Action B.1 provides for placing one inoperable channel in bypass and the other channel in trip within the Completion Time of 1 hour. This Completion Time is sufficient to allow the operator to take all appropriate actions for the failed channels, while ensuring that the risk involved in operating with the failed channels is acceptable. With one channel of protective instrumentation bypassed, the RPS Function is in a two-out-of-three logic; but with another channel failed, the RPS Function may be

operating in a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the RPS Function in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, the reactor will trip.

One instrument channel should be restored to OPERABLE status within 48 hours for reasons similar to those stated under Condition A. After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2 must be placed in trip if more than 48 hours have elapsed since the initial channel failure.

C.1 and C.2

The excore detectors are used to generate the internal ASI used as an input to the TM/LP and APD-High trips. Incore detectors provide a more accurate measurement of ASI. If one or more excore channels cannot be calibrated to match incore detectors, power is restricted or reduced during subsequent operations because of increased uncertainty associated with using uncalibrated excore channels.

The Completion Time of 24 hours is adequate to perform the Surveillance Requirement (SR) while minimizing the risk of operating in an unsafe condition.

D.1, D.2.1, D.2.2.1, and D.2.2.2

Condition D applies to one automatic bypass removal feature inoperable. If the automatic bypass removal feature for any operating bypass channel cannot be restored to OPERABLE status, the associated RPS channel may be considered OPERABLE only if the bypass is not in effect. Otherwise, the affected RPS channel must be declared inoperable, as in Condition A, and the bypass either removed or the automatic bypass removal feature repaired. The Bases for Required Actions and Completion Times are the same as discussed for Condition A.

E.1, E.2.1, and E.2.2

Condition E applies to two inoperable automatic bypass removal features. If the automatic bypass removal features cannot be restored to OPERABLE status, the associated RPS channel may be considered OPERABLE only if the bypasses are not in effect. Otherwise, the affected RPS channels must be declared inoperable, as in Condition B, and the bypasses either removed or the automatic bypass removal features repaired. Also, Required Action E.2.2 provides for the restoration of the one affected RPS channel to OPERABLE status within the rules of Completion Time specified under Condition B. Completion Times are consistent with Condition B.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODEs, even though two automatic bypass removal features are inoperable, with one bistable trip unit bypassed and one tripped. MODE changes in this configuration are allowed to permit maintenance and testing on one of the inoperable automatic bypass removal features. In this configuration, the Function is in a one-out-of-two logic, and the probability of a common cause failure affecting both of the OPERABLE automatic bypass removal features during the 48 hours permitted is remote.

F.1

Condition F is entered when the Required Action and associated Completion Time of Condition A, B, C, D, or E are not met for the APD-High trip and Loss-of-Load trip Functions.

If the Required Actions associated with these Conditions cannot be completed within the required Completion Times, the reactor must be brought to a MODE in which the Required Actions do not apply. The allowed Completion Time of 6 hours to reduce THERMAL POWER to < 15% RTP is reasonable, based on operating experience, to decrease power to < 15% RTP from full power conditions in an orderly manner and without challenging plant systems.

G.1

Condition G is entered when the Required Action and associated Completion Time of Condition A, B, C, D, or E are not met except for the APD-High trip and Loss-of-Load trip Functions.

If the Required Actions associated with these Conditions cannot be completed within the required Completion Times, the reactor must be brought to a MODE in which the Required Actions do not apply. The allowed Completion Time of 6 hours to be in MODE 3 is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The SRs for any particular RPS Function are found in the SR column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one instrument 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 two instrument channels could be an indication of excessive instrument channel drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a qualitative assessment of the instrument channel combined with the instrument channel uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits. CHANNEL CHECKS are performed on the

wide range logarithmic neutron flux monitor for the Rate of Change of Power-High trip Function.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of instrument channel failure. Since the probability of two random failures in redundant channels in any 12-hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of RPS Function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of the channel during normal operational use of the displays.

SR 3.3.1.2

A daily calibration (heat balance) is performed when THERMAL POWER is ≥ 15%. The daily calibration shall consist of adjusting the "nuclear power calibrate" potentiometers to agree with the calorimetric calculation if the absolute difference is > 1.5%. The "ΔT power calibrate" potentiometers are then used to null the "nuclear power-ΔT power" indicators on the RPS Calibration and Indication Panel. Performance of the daily calibration ensures that the two inputs to the Q power measurement are indicating accurately with respect to the much more accurate secondary calorimetric calculation. The heat balance addresses overall gain of the instruments and does not include ASI.

The Frequency of 24 hours is based on plant operating experience and takes into account indications and alarms located in the Control Room to detect deviations in channel outputs. The Frequency is modified by a Note indicating that once the unit reaches 15% RTP, 12 hours is the maximum time allowed for completing this Surveillance. The secondary calorimetric is inaccurate at lower power levels. The 12 hours allows time for plant stabilization, data-taking, and instrument calibration.

A second Note indicates the daily calibration may be suspended during PHYSICS TESTS. This ensures that calibration is proper both preceding and following physics testing at each plateau, recognizing that during testing,

changes in power distribution and RCS temperature may render the calibration inaccurate.

SR 3.3.1.3

It is necessary to calibrate the excore power range channel upper and lower subchannel amplifiers such that the internal ASI used in the TM/LP trip and APD-High trip Functions reflects the true core power distribution as determined by the incore detectors. A Note indicates that once the unit reaches 20% RTP, 12 hours is the maximum time allowed for completion of this Surveillance. The Surveillance is required to be performed prior to operation above 90% RTP. Uncertainties in the excore and incore measurement process make it impractical to calibrate when THERMAL POWER is < 20% RTP. The Completion Time of 12 hours allows time for plant stabilization, data-taking, and instrument calibration. The Frequency requires the SR be performed every 31 days after the initial performance prior to operation above 90% RTP. Requiring the SR prior to operations above 90% RTP is because of the increased uncertainties associated with using uncalibrated excore detectors. If the excore channels are not properly calibrated to agree with the incore detectors, power is restricted during subsequent operations because of increased uncertainty associated with using uncalibrated excore channels. The 31-day Frequency is adequate, based on operating experience of the excore linear amplifiers and the slow burnup of the detectors. The excore readings are a strong function of the power produced in the peripheral fuel bundles and do not represent an integrated reading across the core. Slow changes in neutron flux during the fuel cycle can also be detected at this Frequency.

SR 3.3.1.4

A CHANNEL FUNCTIONAL TEST is performed on each RPS instrument channel, except Loss of Load and Rate of Change of Power, every 92 days to ensure the entire channel will perform its intended function when needed.

In addition to reference voltage power supply tests, the RPS CHANNEL FUNCTIONAL TEST consists of three overlapping tests as described in Reference 1, Section 7.2. These tests

BASES

verify that the RPS is capable of performing its intended function, from bistable input through the RTCBs. They include:

Bistable Tests

The bistable setpoint must be found to trip within the Allowable Values specified in the LCO and left set consistent with the assumptions of Reference 4. As-found values must also be recorded and reviewed for consistency with the assumptions of the frequency extension analysis. The requirements for this review are outlined in Reference 8.

A test signal is substituted as the input in one instrument channel at a time to verify that the bistable trip unit trips within the specified tolerance around the setpoint. This is done with the affected RPS channel bistable trip unit bypassed. Any setpoint adjustment shall be consistent with the assumptions of Reference 4.

Matrix Logic Tests

Matrix logic tests are addressed in LCO 3.3.3. This test is performed one matrix at a time. It verifies that a coincidence in the two instrument channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. This test will detect any short circuits around the bistable contacts in the coincidence logic, such as may be caused by faulty bistable relay or trip bypass contacts.

Trip Path Tests

Trip path logic tests are addressed in LCO 3.3.3. These tests are similar to the matrix logic tests, except that test power is withheld from one matrix relay at a time, allowing the trip path circuit to de-energize, opening the affected set of RTCBs. The RTCBs must then be closed prior to testing the other three trip path circuits, or a reactor trip may result.

The Frequency of 92 days is based on the reliability analysis presented in Reference 6.

SR 3.3.1.5

A CHANNEL CALIBRATION of the excore power range channels every 92 days ensures that the channels are reading accurately and within tolerance. The SR verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the plant-specific SRs.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the Frequency extension analysis. The requirements for this review are outlined in Reference 8.

A Note is added stating that the 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 (Reference 7). Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.2) and the monthly linear subchannel gain check (SR 3.3.1.3). In addition, associated control room indications are continuously monitored by the operators.

The Frequency of 92 days is acceptable, based on plant operating experience, and takes into account indications and alarms available to the operator in the Control Room.

SR 3.3.1.6

A CHANNEL FUNCTIONAL TEST on the Loss of Load, and Rate of Change of Power channels is performed prior to a reactor startup to ensure the entire channel will perform its intended function if required. The Loss of Load sensor cannot be tested during reactor operation without causing reactor trip. The Power Rate of Change-High trip Function is required during startup operation and is bypassed when shut down or > 12% RTP.

BASES

SR 3.3.1.7

Surveillance Requirement 3.3.1.7 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.1.4, except SR 3.3.1.7 is applicable only to Functions with automatic bypass removal features. Proper operation of operating bypasses are critical during plant startup because the bypasses must be in place to allow startup operation and must be removed at the appropriate points during power ascent to enable certain reactor trips. A 24-month SR Frequency is adequate to ensure proper automatic bypass removal feature operation as described in Reference 5. Once the operating bypasses are removed, the bypasses must not fail in such a way that the associated trip Function gets inadvertently bypassed. This feature is verified by the trip Function CHANNEL FUNCTIONAL TEST, SR 3.3.1.4. Therefore, further testing of the automatic bypass removal feature after startup is unnecessary.

SR 3.3.1.8

Surveillance Requirement 3.3.1.8 is the performance of a CHANNEL CALIBRATION every 24 months.

CHANNEL CALIBRATION is a check of the instrument channel, including the sensor. The SR verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument channel drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with Reference 4.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the frequency extension analysis. The requirements for this review are outlined in Reference 6.

The Frequency is based upon the assumption of a 24-month calibration interval for the determination of the magnitude of equipment drift.

The SR is modified by a Note to indicate that the neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift, and because of the

BASES

difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.2) and the monthly linear subchannel gain check (SR 3.3.1.3).

SR 3.3.1.9

This SR ensures that the RPS RESPONSE TIMES are verified to be less than or equal to the maximum values assumed in the safety analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the RTCBs open. Response times are conducted on a 24-month STAGGERED TEST BASIS. Response time testing acceptance criteria are included in Reference 1, Section 7.2. This results in the interval between successive SRs of a given channel of n x 24 months, where n is the number of channels in the function. The Frequency of 24 months is based upon operating experience, which has shown that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. Also, response times cannot be determined at power since equipment operation is required. Testing may be performed in one measurement or in overlapping segments, with verification that all components are tested.

A Note is added to indicate that the neutron detectors are excluded from RPS RESPONSE TIME testing because they are passive devices with minimum drift, and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.3).

REFERENCES

1. Updated Final Safety Analysis Report
2. Title 10 Code of Federal Regulations
3. Institute of Electrical and Electronic Engineers (IEEE) No. 279, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems," August 1968
4. CCNPP Setpoint File

BASES

5. Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated June 5, 1995, "Response to NRC Request for Review & Comment on Review of Preliminary Accident Precursor Analysis of Trip; Loss of 13.8 kV Bus; Short-Term Saltwater Cooling System Unavailability, CCNPP Unit 2"

6. Combustion Engineering Topical Report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" dated June 2, 1986, including Supplement 1, March 3, 1989

7. Letter from Mr. D. G. McDonald (NRC) to Mr. R. E. Denton (BGE), dated October 19, 1995, "Issuance of Amendments for Calvert Cliffs Nuclear Power Plant, Unit No. 1 (TAC No. M92479) and Unit No. 2 (TAC No. M92480)

8. Calvert Cliffs Procedure EN-4-104, "Surveillance Testing"

B 3.3 INSTRUMENTATION

B 3.3.7 Containment Radiation Signal (CRS)

BASES

BACKGROUND

This LCO encompasses CRS actuation, which is a plant-specific instrumentation system that performs an actuation Function required to mitigate offsite dose, but is not otherwise included in LCO 3.3.5 or LCO 3.3.6. This is a non-Nuclear Steam Supply System ESFAS Function that, because of differences in purpose, design, and operating requirements, is not included in LCOs 3.3.5 and 3.3.6.

The CRS provides protection from radioactive contamination in the Containment in the event an irradiated fuel assembly should be severely damaged during handling.

The CRS will detect abnormal amounts of radioactive material in the Containment and will initiate purge valve closure to limit the release of radioactivity to the environment. The containment purge supply and exhaust valves are closed on a CRS when a high radiation level in Containment is detected.

The CRS includes two independent, redundant actuation logic channels. One actuation logic channel ("A" CRS Actuation Logic Channel) secures the containment purge exhaust fan and containment purge supply fan. This actuation logic channel also initiates isolation valve closure. A list of actuated valves and an additional description of the CRS are included in Reference 1, Section 7.3. Both trains of CRS are actuated on a two-out-of-four coincidence from the same four containment radiation sensor channels. Containment purge isolation also occurs on a SIAS. The SIAS is addressed by LCO 3.3.4.

Trip Setpoints and Allowable Values

Trip setpoints used in the sensor modules are based on the analytical limits stated in Reference 1, Chapter 14. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account in the respective analytical limits. To allow for calibration tolerances, instrumentation uncertainties, and sensor channel drift, sensor module trip setpoints are conservatively adjusted with respect to the analytical limits. A detailed

description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in Reference 2. The actual nominal trip setpoint entered into the sensor module is more conservative than that specified by the Allowable Value. One example of such a change in measurement error is drift during the SR interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Sensor channels, measurement channels, sensor modules, and actuation logic are described in the Background for B 3.3.4.

Setpoints in accordance with the Allowable Value will help ensure that 10 CFR Part 100 exposure limits are not violated during a Fuel Handling Accident, providing the plant is operated from within the LCOs at the onset of the Fuel Handling Accident and the equipment functions as designed.

APPLICABLE SAFETY ANALYSES

The CRS satisfies the requirements of 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Only the Allowable Values are specified in the LCO. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the plant-specific setpoint calculations.

Each nominal trip setpoint specified is more conservative than the analytical limit assumed in the Fuel Handling Accident analysis in order to account for instrument uncertainties appropriate to the actuation Function. These uncertainties are defined in Reference 2. A sensor channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

The Bases for the LCO on the CRS are discussed below for each Function:

a. <u>Manual Actuation</u>

The LCO on manual actuation backs up the automatic actuations and ensures operators have the capability to rapidly initiate the CRS Function if any parameter is

BASES

14. Steam Generator Water Level Transmitters

Steam Generator Water Level transmitters are provided to monitor operation of decay heat removal via the steam generators. The Category I indication of steam generator level is the extended startup range level instrumentation. The extended startup range level covers a span of -40 inches to -63 inches (relative to normal operating level), above the lower tubesheet. The measured differential pressure is displayed in inches of water at process conditions of the fluid. Redundant monitoring capability is provided by four transmitters. The uncompensated level signal is input to the plant computer and a control room indicator. Steam generator water level instrumentation consists of two level transmitters.

Operator action is based on the control room indication of steam generator water level. The RCS response during a design basis small break LOCA is dependent on the break size. For a certain range of break sizes, the boiler condenser mode of heat transfer is necessary to remove decay heat. Extended startup range level is a Type A variable because the operator must manually raise and control the steam generator level to establish boiler condenser heat transfer. Feedwater flow is increased until indication is in range.

15. Condensate Storage Tank Level Monitor

Condensate storage tank (CST) level monitoring is provided to ensure water supply for AFW. Condensate Storage Tank 12 provides the ensured safety grade water supply for the AFW System. Inventory in CST 12 is monitored by level indication covering the full range of required usable water level. Condensate storage tank level is displayed on control room indicators and the plant computer. In addition, a control room annunciator alarms on low level.

Condensate storage tank level is considered a Type A variable because the control room meter and annunciator are considered the primary indication used by the Operator. The DBAs that require AFW are the steam line

break and loss of main feedwater. Condensate Storage Tank 12 is the initial source of water for the AFW System. However, as the CST is depleted, manual operator action is necessary to replenish the CST or align suction to the AFW pumps from the hotwell.

16, 17, 18, 19. <u>Core Exit Temperature Monitor</u>

Core Exit Temperature monitors are provided for verification and long-term surveillance of core cooling.

An evaluation was made of the minimum number of valid core exit thermocouples necessary for inadequate core cooling detection. The evaluation determined the reduced complement of core exit thermocouples necessary to detect initial core uncovery and trend the ensuing core heatup. The evaluations account for core nonuniformities, including incore effects of the radial decay power distribution and excore effects of condensate runback in the hot legs and nonuniform inlet temperatures. Based on these evaluations, adequate or inadequate core cooling detection is ensured with two valid core exit thermocouples per quadrant.

The design of the Incore Instrumentation System includes a Type K (chromel alumel) thermocouple within each of the 45 (35 in Unit 2) incore instrument detector assemblies.

The junction of each thermocouple is located more than a foot above the fuel assembly, inside a structure that supports and shields the incore instrument detector assembly string from flow forces in the outlet plenum region. These core exit thermocouples monitor the temperature of the reactor coolant as it exits the fuel assemblies.

The core exit thermocouples have a usable temperature range from 40°F to 2300°F, although accuracy is reduced at temperatures above 1800°F.

B 3.7 PLANT SYSTEMS

B 3.7.8 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The CREVS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity, chemicals, or toxic gas. The CREVS is a shared system providing protection for both Unit 1 and Unit 2.

The CREVS consists of two trains, including redundant outside air intake ducts and redundant emergency recirculation filter trains that recirculate and filter the Control Room air. The CREVS also has shared equipment, including an exhaust-to-atmosphere duct containing redundant isolation valves and a normally closed roof-mounted hatch, an exhaust-to-atmosphere duct from the kitchen and toilet area of the Control Room containing a single isolation valve, and common supply and return ducts in both the standby and emergency recirculation portions of the system. The shared equipment is considered to be a part of each CREVS train. Each CREVS emergency recirculation filter train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodine) and a fan. Instrumentation which actuates the system is addressed in LCOs 3.3.4 and 3.3.8.

The CREVS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation. Actuation of the CREVS ensures the system is in the emergency recirculation mode of operation, ensures the unfiltered outside air intake and unfiltered exhaust-to-atmosphere valves are closed, and aligns the system for emergency recirculation of Control Room air through the redundant trains of HEPA and charcoal filters. The prefilters remove any large particles in the air and any entrained water droplets present to prevent excessive loading of the HEPA filters and charcoal adsorbers. A control room recirculation signal (CRRS) initiates this filtered ventilation of the air supply to the control room.

The air recirculating through the Control Room is continuously monitored by a radiation detector. Detector

output above the setpoint will cause actuation of the CREVS. The CREVS operation in maintaining the Control Room habitable is discussed in Reference 1, Section 9.8.2.3.

The redundant emergency recirculation filter train provides the required filtration should an excessive pressure drop develop across the other filter train. A normally closed hatch and double isolation valves are arranged in series to prevent a breach of isolation from the outside atmosphere, except for the exhaust from the Control Room kitchen and toilet areas. The CREVS is designed in accordance with Seismic Category I requirements.

The CREVS is designed to maintain the Control Room environment for 30 days of continuous occupancy after a DBA without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE SAFETY ANALYSES

The CREVS components are generally arranged in redundant safety-related ventilation trains although some equipment is shared between trains.

The CREVS provides automatic airborne radiological protection for the Control Room Operators, as demonstrated by the Control Room accident dose analyses for the most limiting design basis LOCA fission product release presented in Reference 1, Chapter 14. The CREVS is also credited during a fuel handling accident. The fuel handling accident does not assume a single failure to occur.

The CREVS also provides manually actuated airborne radiological protection for the Control Room operations, for the design basis fuel handling accident presented in Reference 1, Chapter 14.

The worst case single active failure of a component of the CREVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function (except for one valve in the shared duct between the Control Room and the emergency recirculation filter trains).

The CREVS satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

BASES

LCO

The CREVS is required to be OPERABLE to ensure that the Control Room is isolated and at least one emergency recirculation filter train is available, assuming a single failure. Total system failure could result in a Control Room Operator receiving a dose in excess of 5 rem whole body dose in the event of a large radioactive release.

The CREVS is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE. For MODEs 1, 2, 3, and 4, redundancy is required and CREVS is considered OPERABLE when:

a. Both supply fans are OPERABLE;

b. Both recirculation fans are OPERABLE;

c. Both fans included in the emergency recirculation filter trains are OPERABLE;

d. Both HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions;

e. Ductwork, valves, and dampers are OPERABLE, such that air circulation can be maintained; and

f. The Control Room outside air intake can be isolated for the emergency recirculation mode of operation, assuming a single failure.

The LCO is modified by a Note which indicates that only one CREVS redundant component is required to be OPERABLE during movement of irradiated fuel assemblies, when both units are in MODEs 5 or 6, or defueled. Therefore, with both units in other than MODEs 1, 2, 3, or 4, redundancy is not required for movement of irradiated fuel assemblies and CREVS is considered OPERABLE when:

a. One supply fan is OPERABLE;

b. One recirculation fan is OPERABLE;

c. One fan included in the OPERABLE emergency recirculation filter train is OPERABLE;

d. One HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

BASES

e. Associated ductwork, valves, and dampers are OPERABLE, such that air circulation can be maintained and the Control Room can be isolated for the emergency recirculation mode.

When implementing the Note (since redundancy is not required), only one of the two isolation valves in each outside air intake duct is required, and only one of the two isolation valves in the exhaust to atmosphere duct is required. However, the non-operating flow path must be capable of providing isolation of the Control Room from the outside atmosphere.

In addition, the Control Room boundary must be maintained with sufficient integrity to control operator exposure following an accident.

APPLICABILITY

In MODEs 1, 2, 3, and 4, the CREVS must be OPERABLE to limit operator exposure during and following a DBA.

During movement of irradiated fuel assemblies, the CREVS must be OPERABLE to cope with the release from a fuel handling accident. This additional Applicability was added because the CREVS is credited during a fuel handling accident. For clarity, an LCO Note was added to require only one redundant component of CREVS to be OPERABLE when both units are in MODE 5 or 6, or defueled and movement of irradiated fuel is in progress. The fuel handling accident does not assume a single failure to occur. The LCO Note does not identify this requirement on a CREVS train basis so that it is also applicable to redundant components within a train, such as the outside air intake isolation valves. In conjunction with this Note, Action F was added.

ACTIONS

A.1

With one or more ducts with one Control Room outside air intake isolation valve inoperable in MODEs 1, 2, 3, or 4, the OPERABLE Control Room outside air intake valve in each affected duct must be closed immediately. This places the OPERABLE Control Room outside air intake isolation valve in each affected duct in its safety function required position.

B.1

With the toilet area exhaust isolation valve inoperable, action must be taken to restore OPERABLE status within 24 hours. In this Condition, the toilet area exhaust cannot be isolated, therefore, the valve must be restored to OPERABLE status. The 24 hour period allows enough time to repair the valve while limiting the time the toilet area is open to the atmosphere. The 24 hour Completion Time is based on the low probability of a DBA occurring during this time period.

C.1

With one exhaust to atmosphere isolation valve inoperable in MODEs 1, 2, 3, or 4, action must be taken to restore OPERABLE status within seven days. In this Condition, the remaining OPERABLE exhaust to atmosphere isolation valve is adequate to isolate the Control Room. However, the overall reliability is reduced because a single failure in the OPERABLE exhaust to atmosphere isolation valve could result in loss of exhaust to atmosphere isolation valve function. The seven day Completion Time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining exhaust to atmosphere isolation valve to provide the required isolation capability.

D.1

With one CREVS train inoperable for reasons other than Conditions A, B, or C in MODEs 1, 2, 3, or 4, action must be taken to restore OPERABLE status within seven days. In this Condition, the remaining OPERABLE CREVS subsystem is adequate to perform Control Room radiation protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CREVS train could result in loss of CREVS function. The seven day Completion Time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining train to provide the required capability.

E.1 and E.2

If the Required Actions and associated Completion Times of Conditions A, B, C, or D are not met in MODEs 1, 2, 3, or 4,

the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

F.1

Action F provides the actions to be taken when the Required Action and associated Completion Time of Condition B cannot be met, or when two CREVS trains are inoperable for reasons other than Condition B. It requires the immediate suspension of movement of irradiated fuel assemblies. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies to a safe position. Since only one CREVS train must be OPERABLE for movement of irradiated fuel assemblies, the Required Action is applicable only to the "required CREVS."

G.1

If both CREVS trains are inoperable for reasons other than Conditions A, B, or C, or if one or more ducts have two outside air intake isolation valves inoperable, or if two exhaust to atmosphere isolation valves are inoperable, in MODEs 1, 2, 3, or 4, the CREVS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each required CREVS filter train once every month provides an adequate check on this system.

The 31 day Frequency is based on the known reliability of the equipment, and the two filter train redundancy available.

SR 3.7.8.2

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREVS filter tests are in accordance with portions of Reference 2. The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.8.3

This SR verifies each CREVS train starts and operates on an actual or simulated actuation signal (CRRS). This test is conducted on a 24 month Frequency. This Frequency is adequate to ensure the CREVS is capable of starting and operating on an actual or simulated CRRS.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," March 1978

B 3.7 PLANT SYSTEMS

B 3.7.9 Control Room Emergency Temperature System (CRETS)

BASES

BACKGROUND

The CRETS provides temperature control for the Control Room following isolation of the Control Room. The CRETS is a shared system which is supported by the CREVS, since the CREVS must be operating in the emergency recirculation mode for CRETS to perform its safety function.

The CRETS consists of two independent, redundant trains that provide cooling of recirculated Control Room air. Each train consists of cooling coils, instrumentation, and controls to provide for Control Room temperature control. The CRETS is a subsystem providing air temperature control for the Control Room.

The CRETS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation. A single train will provide the required temperature control to maintain the Control Room below 104°F. The CRETS operation to maintain the Control Room temperature is discussed in Reference 1.

APPLICABLE SAFETY ANALYSES

The design basis of the CRETS is to maintain temperature of the Control Room environment throughout 30 days of continuous occupancy.

The CRETS components are arranged in redundant safety-related trains. During emergency operation, the CRETS maintains the temperature below 104°F. A single active failure of a component of the CRETS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for Control Room temperature control. The CRETS is designed in accordance with Seismic Category I requirements. The CRETS is capable of removing sensible and latent heat loads from the Control Room, considering equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

The CRETS satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

BASES

LCO

Two independent and redundant trains of the CRETS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train following isolation of the Control Room. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident requiring isolation of the Control Room.

The CRETS is considered OPERABLE when the individual components that are necessary to maintain the Control Room temperature are OPERABLE. The required components include the cooling coils and associated temperature control instrumentation. In addition, the CRETS must be OPERABLE to the extent that air circulation can be maintained.

For MODEs 1, 2, 3, and 4, redundancy is required and both trains must be OPERABLE. The LCO is modified by a Note which indicates that only one CRETS train is required to be OPERABLE during movement of irradiated fuel assemblies when both units are in MODEs 5 or 6, or defueled. Therefore, with both units in other than MODEs 1, 2, 3, or 4, redundancy is not required for movement of irradiated fuel assemblies and only one CRETS train is required to be OPERABLE.

APPLICABILITY

In MODEs 1, 2, 3, and 4, and during movement of irradiated fuel assemblies, the CRETS must be OPERABLE to ensure that the Control Room temperature will not exceed equipment OPERABILITY requirements following isolation of the Control Room.

The additional Applicability for the movement of irradiated fuel assemblies was added because the CRETS is credited during a fuel handling accident. For clarity, an LCO Note was added to require only one redundant component of CRETS to be OPERABLE when both units are in MODE 5 or 6, or defueled and movement of irradiated fuel is in progress. The fuel handling accident does not assume a single failure to occur. The LCO Note does not identify this requirement on a CRETS train basis so that it is also applicable to redundant components within a train, such as the outside air intake isolation valves. In conjunction with this Note, Action C was added.

ACTIONS

A.1

With one CRETS train inoperable in MODEs 1, 2, 3, or 4, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRETS train is adequate to maintain the Control Room temperature within limits. The 30 day Completion Time is reasonable, based on the low probability of an event occurring requiring Control Room isolation, consideration that the remaining train can provide the required capabilities, and the alternate safety or non-safety-related cooling means that are available.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met in MODEs 1, 2, 3, or 4, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1

During movement of irradiated fuel assemblies, with the required CRETS train inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the Control Room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position. Since only one CRETS train must be OPERABLE for movement of irradiated fuel assemblies, the Required Action is applicable only to the "required CRETS."

D.1

If both CRETS trains are inoperable in MODEs 1, 2, 3, or 4, the CRETS may not be capable of performing the intended function and the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies each required CRETS train has the capability to maintain Control Room temperature ≤ 104°F for ≥ 12 hours in the recirculation mode. During this test, the backup Control Room air conditioner is to be de-energized. This SR consists of a combination of testing. A 24 month Frequency is appropriate, since significant degradation of the CRETS is slow and is not expected over this time period.

REFERENCES

1. UFSAR, Section 9.8.2.3, "Auxiliary Building Ventilating Systems"

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS) and refueling pool during refueling, ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes that have direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the Core Operating Limits Report (COLR). Unit procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control element assemblies and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures. The negative worth of the CEAs may be credited when determining the refueling boron concentrations. Unit procedures maintain the number and position of credited CEAs during fuel handling operations.

Reference 1, Appendix 1C, Criterion 27, requires that two independent reactivity control systems of different design principles be provided. One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling pool. The refueling pool is then flooded with borated water from the refueling water tank into the open reactor vessel by gravity feeding or by the use of the Shutdown Cooling (SDC) System pumps.

BASES

The pumping action of the SDC System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and the refueling pool mix the added concentrated boric acid with the water in the RCS. The SDC System is in operation during refueling [see Limiting Condition of Operation (LCO) 3.9.4 and LCO 3.9.5], to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling pool above the COLR limit.

The COLR includes a MODE 6 temperature limitation of ≤ 140°F. This restriction ensures assumptions made for calculating boron concentration and assumptions made in the boron dilution analysis for MODE 6 are preserved.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the unit refueling procedures that demonstrate the correct fuel loading plan (including full core mapping) ensure the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer tube, the refueling pool, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Reference 1, Chapter 14). A detailed discussion of this event is provided in B 3.1.1.

The RCS boron concentration satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within the Containment Structure, a release of fission product radioactivity within the Containment Structure will be restricted from escaping to the environment when the LCO requirements are met. In MODEs 1, 2, 3, and 4, this is accomplished by maintaining Containment OPERABLE, as described in LCO 3.6.1. In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment atmosphere from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no design basis accident potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The Containment Structure serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR Part 100. Additionally, the Containment Structure provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch opening provides a means for moving large equipment and components into and out of the Containment Structure. During CORE ALTERATIONS or movement of irradiated fuel assemblies within Containment, the equipment hatch must be held in place by at least four bolts or the containment outage door must be capable of being closed. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are part of the containment pressure boundary, provide a means for personnel access during MODEs 1, 2, 3, and 4 operation in accordance with LCO 3.6.2. Each air lock has a door at both ends. The

doors are normally interlocked to prevent simultaneous opening when Containment OPERABILITY is required.

In other situations, the potential for containment pressurization as a result of an accident is not present, therefore, less stringent requirements are needed to isolate the containment atmosphere from the outside atmosphere. Both containment personnel air lock doors and the containment outage door may be open during the movement of irradiated fuel assemblies in containment and during CORE ALTERATIONS; provided one air lock door and the containment outage door are OPERABLE, the plant is in MODE 6 with at least 23 ft of water above the fuel, and a designated individual is continuously available to close each door. The designated individuals must be stationed at the Auxiliary Building side of the outer air lock door and at the outside of the containment equipment hatch. OPERABILITY of a containment personnel air lock door and the containment outage door requires that the door is capable of being closed, that the door is unblocked, and no cables or hoses are run through the doorway. Containment outage door grating may be installed if the grating can be removed with the use of a forklift and the door closed within 30 minutes. During CORE ALTERATIONS or movement of irradiated fuel assemblies in containment, the requirement for at least 23 ft of water above the fuel, ensures that there is sufficient time to close the personnel air lock and the containment outage door following a loss of SDC before boiling occurs and minimizes activity release after a fuel handling accident. The personnel air lock door and the containment outage door may be operated independently of each other (i.e., they do not have to be open or shut at the same time).

The requirements on containment penetration closure, ensure that a release of fission product radioactivity within the Containment Structure will be restricted to within regulatory limits.

The Containment Purge Valve Isolation System, for the purposes of compliance with LCO 3.9.3, item d.2, includes a 48 inch purge penetration and a 48 inch exhaust penetration. For the purposes of compliance with LCO 3.9.3, the

containment vent isolation valves are not considered part of the Containment Purge Valve Isolation System since they may not be capable of being closed automatically. The containment vent, includes a four inch purge penetration and a four inch exhaust penetration. During MODEs 1, 2, 3, and 4, the normal purge and exhaust penetrations are isolated (via a blind flange, if installed or by the purge valves). The containment vent valves can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System. Neither of the subsystems is subject to a Specification in MODE 5.

In MODE 6, large air exchanges are desired to conduct refueling operations. The normal 48 inch purge system is used for this purpose and all valves are closed by the Engineered Safety Features Actuation System in accordance with LCO 3.3.7.

The containment vent isolation valves are required to be closed during CORE ALTERATIONS or movement of irradiated fuel within Containment. These valves are connected to the penetration room Technical Specification emergency air cleanup systems, which exhaust to the outside atmosphere through high efficiency particulate air and charcoal filters.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved in accordance with appropriate American Society of Mechanical Engineers / American National Standards Institute Codes, and may include use of a material that can provide a temporary ventilation barrier for the other containment penetrations during fuel movements.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within the Containment Structure, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Reference 1). The

fuel handling accident, described in Reference 1, includes dropping a single irradiated fuel assembly which would then rotate to a horizontal position, strike a protruding structure, and rupture the fuel pins. The requirements of LCO 3.9.6, and the minimum decay time of 100 hours prior to CORE ALTERATIONS, ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are within the acceptance limits given in Reference 1.

Containment penetrations satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

This LCO limits the consequences of a fuel handling accident in Containment Structure, by limiting the potential escape paths for fission product radioactivity released within Containment. The LCO requires any penetration providing direct access from the Containment Structure atmosphere to the outside atmosphere (including the containment vent isolation valves) to be closed, except for the OPERABLE containment purge and exhaust penetrations, the containment personnel air locks and the containment outage door. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge Valve Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure times specified in the UFSAR can be achieved, and therefore meet the assumptions used in the safety analysis to ensure releases through the valves are terminated, such that the radiological doses are within the acceptance limit.

Both containment personnel air lock doors and the containment outage door may be open under administrative controls during movement of irradiated fuel in Containment and during CORE ALTERATIONS provided that one OPERABLE personnel air lock door and an OPERABLE containment outage door are capable of being closed in the event of a fuel handling accident. The administrative controls consist of designated individuals available immediately outside the personnel air lock and the containment outage door to close the OPERABLE doors. Should a fuel handling accident occur inside the Containment Structure, one personnel air lock

door and the containment outage door will be closed following an evacuation of the Containment Structure.

The LCO is modified by a Note which allows the emergency air lock temporary closure device to replace an emergency air lock door. The temporary closure device provides an adequate barrier to shield the environment from the containment atmosphere in case of a design basis event that does not create a pressure increase inside Containment.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within the Containment Structure because this is when there is a potential for a fuel handling accident. In MODEs 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODEs 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within the Containment Structure are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

With the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, (including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open) the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within the Containment Structure. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This SR demonstrates that each of the containment penetrations required to be in its closed position, is in that position. The surveillance test on the open purge and exhaust valves will demonstrate that the valves are not

blocked from closing. Also, the surveillance test will demonstrate that each purge and exhaust valve operator has motive power, which will ensure each valve is capable of being closed by an OPERABLE automatic Containment Purge Valve Isolation System.

The surveillance test is performed every seven days during CORE ALTERATIONS or movement of irradiated fuel assemblies within the Containment Structure. The surveillance test interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance test before the start of refueling operations will provide two or three verifications during the applicable period for this LCO. As such, this SR ensures that a postulated fuel handling accident, that releases fission product radioactivity within the Containment Structure, will not result in a release of fission product radioactivity to the environment in excess of those described in Reference 1.

SR 3.9.3.2

This SR demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The once each refueling outage Frequency, maintains consistency with other similar Engineered Safety Features Actuation System instrumentation and valve testing requirements. However, in order to ensure the SR Frequency is satisfied, this surveillance test is typically performed once per refueling outage prior to the start of CORE ALTERATIONS or movement of irradiated fuel assemblies within Containment. In LCO 3.3.7, the Containment Radiation Signal System requires a CHANNEL CHECK every 12 hours and a CHANNEL FUNCTIONAL TEST every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 24 months during refueling on a STAGGERED TEST BASIS. Surveillance Requirement 3.6.3.4 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These surveillance tests performed during MODE 6 will ensure that the valves are capable of

BASES

closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the Containment Structure.

REFERENCES

1. UFSAR, Section 14.18, "Fuel Handling Incident"

concentration, CORE ALTERATIONS are suspended and all containment penetrations must be in the status described in LCO 3.9.3. This allowance is necessary to perform required maintenance and testing.

APPLICABILITY

One SDC loop must be in operation in MODE 6, with the water level ≥ 23 ft above the top of the irradiated fuel assemblies seated in the reactor vessel, to provide decay heat removal. The 23 ft level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6.

Requirements for the SDC System in other MODEs are covered by LCOs in Section 3.4 and Section 3.5. Shutdown cooling loop requirements in MODE 6, with the water level < 23 ft above the top of the irradiated fuel assemblies seated in the reactor vessel, are located in LCO 3.9.5.

ACTIONS

Shutdown cooling loop requirements are met by having one SDC loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If one required SDC loop is inoperable or not in operation, action shall be immediately initiated and continued until the SDC loop is restored to OPERABLE status and to operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

A.2

If SDC loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur through the addition of water with a lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately. In addition, to ensure compliance with the action is maintained, the charging pumps shall be de-energized and charging flow paths closed as part of Required Action A.2.

BASES

A.3

If SDC loop requirements are not met, actions shall be taken immediately to suspend loading irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the irradiated fuel assemblies seated in the reactor vessel provides an adequate available heat sink. Suspending any operation that would increase the decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.4

If SDC loop requirements are not met, all containment penetrations to the outside atmosphere must be closed to prevent fission products, if released by a loss of decay heat event, from escaping the Containment Structure. The four hour Completion Time allows fixing most SDC problems without incurring the additional action of violating the containment atmosphere. The emergency air lock temporary closure device cannot be credited for containment closure for a loss of shutdown cooling event. At least one door in the emergency air lock must be closed to satisfy this action statement.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This SR demonstrates that the SDC loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability, and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the Control Room for monitoring the SDC System.

REFERENCES

1. UFSAR, Section 9.2, "Shutdown Cooling System"

BASES

seated in the reactor vessel, the Applicability will change to that of LCO 3.9.4, and only one SDC loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no SDC loop is in operation or no SDC loops are OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by the addition of water with lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately. In addition, to ensure compliance with the action is maintained, the charging pumps shall be de-energized and charging flow paths closed as part of Required Action B.1.

B.2

If no SDC loop is in operation or no SDC loops are OPERABLE, action shall be initiated immediately and continued without interruption to restore one SDC loop to OPERABLE status and operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE SDC loops and one operating SDC loop should be accomplished expeditiously.

B.3

If no SDC loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within four hours. With the SDC loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded. The emergency air lock temporary closure device cannot be credited for containment closure for a loss of shutdown cooling event. At least one door in the emergency air lock must e closed to satisfy this action statement.

The Completion Time of four hours is reasonable, based on the low probability of the coolant boiling in that time.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This SR demonstrates that one SDC loop is operating and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. This SR also demonstrates that the other SDC loop is OPERABLE.

In addition, during operation of the SDC loop with the water level in the vicinity of the reactor vessel nozzles, the SDC loop flow rate determination must also consider the SDC pump suction requirements. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the SDC System in the Control Room.

Verification that the required loops are OPERABLE and in operation ensures that loops can be placed in operation as needed, to maintain decay heat and retain forced circulation. The Frequency of 12 hours is considered reasonable, since other administrative controls are available and have proven to be acceptable by operating experience.

SR 3.9.5.2

This SR demonstrates that the SDC loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available for the operator in the Control Room for monitoring the SDC System.

SR 3.9.5.3

Verification that the required pump and valves are OPERABLE ensures that an additional SDC loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump and valves. The Frequency of seven days

PAGE REPLACEMENT INSTRUCTIONS
Calvert Cliffs Nuclear Power Plant

Technical Specification Bases - Revision 12

Remove and Discard	**Insert**
List of Effective Pages	
LEP-1 through LEP-5	LEP-1 through LEP-5
List of Revisions	
LOR-1	LOR-1
Technical Specification Bases Pages	
B 3.2.2-3 through B.3.2.2-5	B 3.2.2-3 through B.3.2.2-6
B 3.2.3-3 through B 3.2.3-5	B 3.2.3-3 through B 3.2.3-5
B 3.3.1-5 through B 3.3.1-14	B 3.3.1-5 through B 3.3.1-14
B 3.3.1-33 and B 3.3.1-34	B 3.3.1-33 and B 3.3.1-34
B 3.3.4-21 and B.3.3.4-23	B 3.3.4-21 and B.3.3.4-23
B 3.3.6-3 through B 3.3.6-8	B 3.3.6-3 through B 3.3.6-8
B 3.4.11-1 through B 3.4.11-7	B 3.4.11-1 through B 3.4.11-7
B 3.6.1-3 and B 3.6.1-4	B 3.6.1-3 and B 3.6.1-4
B 3.7.1-3 through B 3.7.1-5	B 3.7.1-3 through B 3.7.1-5
B 3.7.3-3 and B 3.7.3-10	B 3.7.3-3 and B 3.7.3-10
B 3.7.7-1 and B 3.7.7-4	B 3.7.7-1 and B 3.7.7-4
B 3.8.1-1 and B 3.8.1-2	B 3.8.1-1 and B 3.8.1-2

November 13, 2001

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

LEP-1	Rev. 12	B 3.1.2-2	Rev. 2	B 3.2.1-2	Rev. 11
LEP-2	Rev. 12	B 3.1.2-3	Rev. 2	B 3.2.1-3	Rev. 11
LEP-3	Rev. 12	B 3.1.2-4	Rev. 2	B 3.2.1-4	Rev. 11
LEP-4	Rev. 12	B 3.1.2-5	Rev. 2	B 3.2.1-5	Rev. 11
LEP-5	Rev. 12	B 3.1.2-6	Rev. 3	B 3.2.1-6	Rev. 11
i	Rev. 2	B 3.1.3-1	Rev. 2	B 3.2.1-7	Rev. 11
ii	Rev. 2	B 3.1.3-2	Rev. 2	B 3.2.2-1	Rev. 2
iii	Rev. 2	B 3.1.3-3	Rev. 2	B 3.2.2-2	Rev. 8
iv	Rev. 2	B 3.1.3-4	Rev. 2	B 3.2.2-3	Rev. 8
B 2.1.1-1	Rev. 2	B 3.1.3-5	Rev. 2	B 3.2.2-4	Rev. 12
B 2.1.1-2	Rev. 2	B 3.1.4-1	Rev. 2	B 3.2.2-5	Rev. 12
B 2.1.1-3	Rev. 2	B 3.1.4-2	Rev. 2	B 3.2.2-6	Rev. 12
B 2.1.1-4	Rev. 2	B 3.1.4-3	Rev. 2	B 3.2.3-1	Rev. 2
B 2.1.2-1	Rev. 2	B 3.1.4-4	Rev. 2	B 3.2.3-2	Rev. 8
B 2.1.2-2	Rev. 2	B 3.1.4-5	Rev. 2	B 3.2.3-3	Rev. 8
B 2.1.2-3	Rev. 2	B 3.1.4-6	Rev. 2	B 3.2.3-4	Rev. 12
B 2.1.2-4	Rev. 2	B 3.1.4-7	Rev. 2	B 3.2.3-5	Rev. 12
B 3.0-1	Rev. 2	B 3.1.4-8	Rev. 2	B 3.2.4-1	Rev. 2
B 3.0-2	Rev. 2	B 3.1.4-9	Rev. 2	B 3.2.4-2	Rev. 8
B 3.0-3	Rev. 2	B 3.1.4-10	Rev. 2	B 3.2.4-3	Rev. 8
B 3.0-4	Rev. 2	B 3.1.5-1	Rev. 2	B 3.2.4-4	Rev. 8
B 3.0-5	Rev. 2	B 3.1.5-2	Rev. 2	B 3.2.4-5	Rev. 8
B 3.0-6	Rev. 2	B 3.1.5-3	Rev. 2	B 3.2.5-1	Rev. 2
B 3.0-7	Rev. 2	B 3.1.5-4	Rev. 2	B 3.2.5-2	Rev. 11
B 3.0-8	Rev. 2	B 3.1.5-5	Rev. 2	B 3.2.5-3	Rev. 11
B 3.0-9	Rev. 2	B 3.1.6-1	Rev. 2	B 3.2.5-4	Rev. 11
B 3.0-10	Rev. 2	B 3.1.6-2	Rev. 2	B 3.2.5-5	Rev. 11
B 3.0-11	Rev. 2	B 3.1.6-3	Rev. 2	B 3.2.5-6	Rev. 11
B 3.0-12	Rev. 2	B 3.1.6-4	Rev. 2	B 3.3.1-1	Rev. 2
B 3.0-13	Rev. 2	B 3.1.6-5	Rev. 2	B 3.3.1-2	Rev. 2
B 3.0-14	Rev. 2	B 3.1.6-6	Rev. 2	B 3.3.1-3	Rev. 2
B 3.0-15	Rev. 2	B 3.1.6-7	Rev. 2	B 3.3.1-4	Rev. 2
B 3.0-16	Rev. 2	B 3.1.7-1	Rev. 2	B 3.3.1-5	Rev. 2
B 3.0-17	Rev. 2	B 3.1.7-2	Rev. 2	B 3.3.1-6	Rev. 12
B 3.0-18	Rev. 2	B 3.1.7-3	Rev. 5	B 3.3.1-7	Rev. 12
B 3.1.1-1	Rev. 2	B 3.1.7-4	Rev. 2	B 3.3.1-8	Rev. 12
B 3.1.1-2	Rev. 2	B 3.1.7-5	Rev. 11	B 3.3.1-9	Rev. 12
B 3.1.1-3	Rev. 2	B 3.1.8-1	Rev. 2	B 3.3.1-10	Rev. 12
B 3.1.1-4	Rev. 2	B 3.1.8-2	Rev. 2	B 3.3.1-11	Rev. 5
B 3.1.1-5	Rev. 2	B 3.1.8-3	Rev. 2	B 3.3.1-12	Rev. 2
B 3.1.1-6	Rev. 2	B 3.1.8-4	Rev. 2	B 3.3.1-13	Rev. 12
B 3.1.1-7	Rev. 2	B 3.1.8-5	Rev. 2	B 3.3.1-14	Rev. 11
B 3.1.2-1	Rev. 2	B 3.2.1-1	Rev. 2	B 3.3.1-15	Rev. 11

TECHNICAL SPECIFICATIONS BASES
LIST OF EFFECTIVE PAGES

B 3.3.1-16	Rev. 11	B 3.3.4-2	Rev. 2	B 3.3.6-8	Rev. 8
B 3.3.1-17	Rev. 11	B 3.3.4-3	Rev. 2	B 3.3.7-1	Rev. 11
B 3.3.1-18	Rev. 11	B 3.3.4-4	Rev. 2	B 3.3.7-2	Rev. 2
B 3.3.1-19	Rev. 11	B 3.3.4-5	Rev. 2	B 3.3.7-3	Rev. 2
B 3.3.1-20	Rev. 11	B 3.3.4-6	Rev. 2	B 3.3.7-4	Rev. 2
B 3.3.1-21	Rev. 11	B 3.3.4-7	Rev. 2	B 3.3.7-5	Rev. 2
B 3.3.1-22	Rev. 11	B 3.3.4-8	Rev. 2	B 3.3.7-6	Rev. 2
B 3.3.1-23	Rev. 11	B 3.3.4-9	Rev. 2	B 3.3.7-7	Rev. 2
B 3.3.1-24	Rev. 11	B 3.3.4-10	Rev. 2	B 3.3.8-1	Rev. 8
B 3.3.1-25	Rev. 11	B 3.3.4-11	Rev. 2	B 3.3.8-2	Rev. 2
B 3.3.1-26	Rev. 11	B 3.3.4-12	Rev. 2	B 3.3.8-3	Rev. 2
B 3.3.1-27	Rev. 11	B 3.3.4-13	Rev. 2	B 3.3.8-4	Rev. 2
B 3.3.1-28	Rev. 11	B 3.3.4-14	Rev. 2	B 3.3.9-1	Rev. 2
B 3.3.1-29	Rev. 11	B 3.3.4-15	Rev. 2	B 3.3.9-2	Rev. 2
B 3.3.1-30	Rev. 11	B 3.3.4-16	Rev. 2	B 3.3.9-3	Rev. 2
B 3.3.1-31	Rev. 11	B 3.3.4-17	Rev. 5	B 3.3.9-4	Rev. 2
B 3.3.1-32	Rev. 11	B 3.3.4-18	Rev. 2	B 3.3.9-5	Rev. 2
B 3.3.1-33	Rev. 12	B 3.3.4-19	Rev. 2	B 3.3.9-6	Rev. 2
B 3.3.1-34	Rev. 12	B 3.3.4-20	Rev. 2	B 3.3.9-7	Rev. 2
B 3.3.1-35	Rev. 2	B 3.3.4-21	Rev. 2	B 3.3.9-8	Rev. 3
B 3.3.2-1	Rev. 2	B 3.3.4-22	Rev. 12	B 3.3.10-1	Rev. 2
B 3.3.2-2	Rev. 2	B 3.3.4-23	Rev. 12	B 3.3.10-2	Rev. 2
B 3.3.2-3	Rev. 2	B 3.3.5-1	Rev. 2	B 3.3.10-3	Rev. 2
B 3.3.2-4	Rev. 2	B 3.3.5-2	Rev. 2	B 3.3.10-4	Rev. 2
B 3.3.2-5	Rev. 2	B 3.3.5-3	Rev. 2	B 3.3.10-5	Rev. 2
B 3.3.2-6	Rev. 2	B 3.3.5-4	Rev. 2	B 3.3.10-6	Rev. 2
B 3.3.2-7	Rev. 2	B 3.3.5-5	Rev. 2	B 3.3.10-7	Rev. 2
B 3.3.2-8	Rev. 2	B 3.3.5-6	Rev. 2	B 3.3.10-8	Rev. 2
B 3.3.2-9	Rev. 2	B 3.3.5-7	Rev. 2	B 3.3.10-9	Rev. 8
B 3.3.2-10	Rev. 2	B 3.3.5-8	Rev. 2	B 3.3.10-10	Rev. 11
B 3.3.3-1	Rev. 2	B 3.3.5-9	Rev. 2	B 3.3.10-11	Rev. 2
B 3.3.3-2	Rev. 2	B 3.3.5-10	Rev. 2	B 3.3.10-12	Rev. 5
B 3.3.3-3	Rev. 2	B 3.3.5-11	Rev. 2	B 3.3.10-13	Rev. 5
B 3.3.3-4	Rev. 2	B 3.3.5-12	Rev. 2	B 3.3.10-14	Rev. 2
B 3.3.3-5	Rev. 2	B 3.3.5-13	Rev. 2	B 3.3.10-15	Rev. 2
B 3.3.3-6	Rev. 2	B 3.3.5-14	Rev. 2	B 3.3.10-16	Rev. 2
B 3.3.3-7	Rev. 2	B 3.3.6-1	Rev. 2	B 3.3.10-17	Rev. 2
B 3.3.3-8	Rev. 2	B 3.3.6-2	Rev. 2	B 3.3.11-1	Rev. 2
B 3.3.3-9	Rev. 2	B 3.3.6-3	Rev. 2	B 3.3.11-2	Rev. 2
B 3.3.3-10	Rev. 2	B 3.3.6-4	Rev. 12	B 3.3.11-3	Rev. 2
B 3.3.3-11	Rev. 2	B 3.3.6-5	Rev. 2	B 3.3.11-4	Rev. 2
B 3.3.3-12	Rev. 2	B 3.3.6-6	Rev. 2	B 3.3.11-5	Rev. 2
B 3.3.4-1	Rev. 2	B 3.3.6-7	Rev. 12	B 3.3.12-1	Rev. 2

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

B 3.6.2-4	Rev. 2	B 3.7.1-4	Rev. 12	B 3.7.9-2	Rev. 11
B 3.6.2-5	Rev. 2	B 3.7.1-5	Rev. 12	B 3.7.9-3	Rev. 11
B 3.6.2-6	Rev. 2	B 3.7.2-1	Rev. 2	B 3.7.9-4	Rev. 11
B 3.6.2-7	Rev. 2	B 3.7.2-2	Rev. 2	B 3.7.10-1	Rev. 9
B 3.6.2-8	Rev. 2	B 3.7.2-3	Rev. 2	B 3.7.10-2	Rev. 2
B 3.6.3-1	Rev. 2	B 3.7.2-4	Rev. 2	B 3.7.10-3	Rev. 2
B 3.6.3-2	Rev. 2	B 3.7.2-5	Rev. 2	B 3.7.11-1	Rev. 2
B 3.6.3-3	Rev. 2	B 3.7.3-1	Rev. 2	B 3.7.11-2	Rev. 2
B 3.6.3-4	Rev. 2	B 3.7.3-2	Rev. 2	B 3.7.11-3	Rev. 2
B 3.6.3-5	Rev. 2	B 3.7.3-3	Rev. 12	B 3.7.11-4	Rev. 2
B 3.6.3-6	Rev. 2	B 3.7.3-4	Rev. 12	B 3.7.12-1	Rev. 2
B 3.6.3-7	Rev. 2	B 3.7.3-5	Rev. 12	B 3.7.12-2	Rev. 2
B 3.6.3-8	Rev. 2	B 3.7.3-6	Rev. 12	B 3.7.12-3	Rev. 2
B 3.6.3-9	Rev. 2	B 3.7.3-7	Rev. 12	B 3.7.12-4	Rev. 2
B 3.6.3-10	Rev. 2	B 3.7.3-8	Rev. 12	B 3.7.13-1	Rev. 8
B 3.6.4-1	Rev. 2	B 3.7.3-9	Rev. 12	B 3.7.13-2	Rev. 8
B 3.6.4-2	Rev. 2	B 3.7.3-10	Rev. 12	B 3.7.13-3	Rev. 8
B 3.6.4-3	Rev. 2	B 3.7.4-1	Rev. 2	B 3.7.14-1	Rev. 2
B 3.6.5-1	Rev. 2	B 3.7.4-2	Rev. 8	B 3.7.14-2	Rev. 2
B 3.6.5-2	Rev. 2	B 3.7.4-3	Rev. 2	B 3.7.14-3	Rev. 2
B 3.6.5-3	Rev. 3	B 3.7.4-4	Rev. 2	B 3.7.15-1	Rev. 2
B 3.6.6-1	Rev. 2	B 3.7.5-1	Rev. 2	B 3.7.15-2	Rev. 2
B 3.6.6-2	Rev. 2	B 3.7.5-2	Rev. 2	B 3.7.15-3	Rev. 2
B 3.6.6-3	Rev. 2	B 3.7.5-3	Rev. 2	B 3.7.15-4	Rev. 2
B 3.6.6-4	Rev. 5	B 3.7.5-4	Rev. 2	B 3.8.1-1	Rev. 5
B 3.6.6-5	Rev. 2	B 3.7.5-5	Rev. 2	B 3.8.1-2	Rev. 12
B 3.6.6-6	Rev. 2	B 3.7.6-1	Rev. 5	B 3.8.1-3	Rev. 2
B 3.6.6-7	Rev. 2	B 3.7.6-2	Rev. 2	B 3.8.1-4	Rev. 10
B 3.6.6-8	Rev. 2	B 3.7.6-3	Rev. 5	B 3.8.1-5	Rev. 7
B 3.6.6-9	Rev. 2	B 3.7.6-4	Rev. 5	B 3.8.1-6	Rev. 7
B 3.6.7-1	Rev. 2	B 3.7.6-5	Rev. 5	B 3.8.1-7	Rev. 3
B 3.6.7-2	Rev. 2	B 3.7.7-1	Rev. 5	B 3.8.1-8	Rev. 3
B 3.6.7-3	Rev. 2	B 3.7.7-2	Rev. 12	B 3.8.1-9	Rev. 3
B 3.6.7-4	Rev. 2	B 3.7.7-3	Rev. 2	B 3.8.1-10	Rev. 3
B 3.6.7-5	Rev. 2	B 3.7.7-4	Rev. 12	B 3.8.1-11	Rev. 3
B 3.6.7-6	Rev. 2	B 3.7.8-1	Rev. 8	B 3.8.1-12	Rev. 3
B 3.6.8-1	Rev. 2	B 3.7.8-2	Rev. 11	B 3.8.1-13	Rev. 3
B 3.6.8-2	Rev. 2	B 3.7.8-3	Rev. 11	B 3.8.1-14	Rev. 3
B 3.6.8-3	Rev. 2	B 3.7.8-4	Rev. 11	B 3.8.1-15	Rev. 3
B 3.6.8-4	Rev. 2	B 3.7.8-5	Rev. 11	B 3.8.1-16	Rev. 3
B 3.7.1-1	Rev. 2	B 3.7.8-6	Rev. 11	B 3.8.1-17	Rev. 3
B 3.7.1-2	Rev. 9	B 3.7.8-7	Rev. 11	B 3.8.1-18	Rev. 3
B 3.7.1-3	Rev. 9	B 3.7.9-1	Rev. 2	B 3.8.1-19	Rev. 3

TECHNICAL SPECIFICATIONS BASES
LIST OF EFFECTIVE PAGES

TECHNICAL SPECIFICATION BASES

LIST OF REVISIONS AND ISSUE DATES

Rev.	Date Issued	Date to NRC
0		May 4, 1998
1	August 28, 1998	October 30, 1998
2	August 28, 1998	October 30, 1998
3	October 28, 1998	October 30, 1998
4	March 16, 1999	October 18, 1999
5	October 18, 1999	October 18, 1999
6	April 14, 2000	October 24, 2000
7	May 18, 2000	October 24, 2000
8	June 29, 2000	October 24, 2000
9	October 24, 2000	October 24, 2000
10	February 1, 2001	November 13, 2001
11	March 22, 2001	November 13, 2001
12	November 13, 2001	November 13, 2001

Alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Reference 2);

b. During a LOFA, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected CEA accident, the energy input to the fuel must not exceed the accepted limits (Reference 1, Section 14.13); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck, fully withdrawn (Reference 1, Appendix 1C, Criterion 29).

The power density at any point in the core must be limited to maintain the fuel design criteria (Reference 2). This limiting is accomplished by maintaining the power distribution and reactor coolant conditions such that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Reference 1, Chapter 14), with due regard for the correlations between measured quantities, the power distribution, and the uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum LHGR so that the peak cladding temperature does not exceed 2200°F (Reference 2). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy-water reaction.

The LCOs governing LHR, ASI, and the RCS ensure that these criteria are met as long as the core is operated within the ASI, F^T_{xy}, F^T_r, and T_q limits specified in the COLR. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

BASES

Fuel cladding damage does not normally occur while at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, should an accident or AOO occur from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHR.

F^T_{xy} satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNB ratio operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limitation on LHR ensures that in the event of a LOCA the peak temperature of the fuel cladding does not exceed 2200°F.

APPLICABILITY

In MODE 1, power distribution must be maintained within the limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In other MODEs, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution.

ACTIONS

A.1

The limitations on F^T_{xy} provided in the COLR ensure that the assumptions used in the analysis for establishing the LHR, LCO, and LSSS remain valid during operation at the various allowable CEA group insertion limits. If F^T_{xy} exceeds its basic limitation (F^T_{xy} > all rods out, full power limit), six hours is provided to restore F^T_{xy} to within limits. The combination of THERMAL POWER and F^T_{xy} must be brought to within the limits established in the COLR and the CEAs must be withdrawn to or above the long-term steady state insertions limits of Technical Specification 3.1.6. Six hours to return F^T_{xy} to within its limit is reasonable

BASES

and is sufficiently short to minimize the time F^T_{xy} is not within limits.

B.1

If F^T_{xy} cannot be returned to within its limit, THERMAL POWER must be reduced to MODE 2. A change to MODE 2 provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition. 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.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The periodic SR to determine the calculated F^T_{xy} ensures that F^T_{xy} remains within the range assumed in the analysis throughout the fuel cycle. Determining the measured F^T_{xy} after each fuel loading prior to the reactor exceeding 70% RTP ensures that the core is properly loaded.

Performance of the SR every 31 days of accumulated operation in MODE 1 provides reasonable assurance that unacceptable changes in the F^T_{xy} are promptly detected.

The power distribution map can only be obtained after THERMAL POWER exceeds 20% RTP because the incore detectors are not reliable below 20% RTP.

The SR is modified by a Note that requires the incore detectors to be used to determine F^T_{xy} by using them to obtain a power distribution map with all full length CEAs above the long term steady state insertion limits, as specified in the COLR. This determination is limited to core planes between 15% and 85% of full core height inclusive and still exclude regions influenced by grid effects.

BASES

REFERENCES	1.	UFSAR
	2.	10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

b. During a LOFA, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected CEA accident, the energy input to the fuel must not exceed the accepted limits (Reference 1, Section 14.13); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Reference 1, Appendix 1C, Criterion 29).

The power density at any point in the core must be limited to maintain the fuel design criteria (Reference 2). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Reference 1, Chapter 14), with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum LHGR so that the peak cladding temperature does not exceed 2200°F (Reference 2). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy-water reaction.

The LCOs governing LHR, ASI, and the RCS ensure that these criteria are met as long as the core is operated within the ASI, F_{xy}^T, and F_r^T limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three-dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analysis.

Fuel cladding damage does not normally occur while at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, if an accident or AOO occurs from initial conditions outside

the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution cause increased power peaking and correspondingly increased local LHR.

F_r^T satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The LCO limits for power distribution are based on correlations between power peaking and measured variables used as inputs to LHR and DNB ratio operating limits. The LCO limits for power distribution, except T_q, are provided in the COLR. The limitation on the LHR ensures that, in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

APPLICABILITY

In MODE 1, power distribution must be maintained within the limits assumed in the accident analysis to ensure that fuel damage does not result following an AOO. In other MODEs, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution.

ACTIONS

A.1

The limitations on F_r^T provided in the COLR ensure that the assumptions used in the analysis for establishing the ASI, LCO, and LSSS remain valid during operation at the various allowable CEA group insertion limits. If F_r^T exceeds its basic limitation (F_r^T > all rods out, full power limit), 6 hours is provided to restore F_r^T to within limits. The combination of THERMAL POWER and F_r^T must be brought to within the limits established in the COLR and the CEAs must be withdrawn to or above the long-term steady state insertions limits of Technical Specification 3.1.6. Six hours to return F_r^T to within its limits is reasonable and is sufficiently short to minimize the time F_r^T is not within limits.

B.1

If F_r^T cannot be returned to within its limit, THERMAL POWER must be reduced to MODE 2. A change to MODE 2 provides

BASES

reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition. 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.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The periodic SR to determine the calculated F_r^T ensures that F_r^T remains within the range assumed in the analysis throughout the fuel cycle. Determining the measured F_r^T once after each fuel loading prior to exceeding 70% RTP ensures that the core is properly loaded.

Performance of the SR every 31 days of accumulated operation in MODE 1 provides reasonable assurance that unacceptable changes in the F_r^T are promptly detected.

The power distribution map can only be obtained after THERMAL POWER exceeds 20% RTP because the incore detectors are not reliable below 20% RTP.

The SR is modified by a Note that requires the incore detectors to be used to determine F_r^T by using them to obtain a power distribution map with all full length CEAs above the long-term steady state insertion limits, as specified in the COLR.

REFERENCES

1. UFSAR
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

inputs and represents a minimum acceptable RCS pressure to be compared to actual RCS pressure in the TM/LP trip unit.

Steam generator pressure is also an indirect input to the TM/LP trip via the ASGT. This Function provides a reactor trip when the secondary pressure in either steam generator exceeds that of the other generator by greater than a fixed amount. The trip is implemented by biasing the TM/LP trip setpoint upward so as to ensure TM/LP trip if an ASGT is detected.

- APD-High Trip

  Q power and subchannel deviation are inputs to the APD trip. The APD trip setpoint is a function of Q power, being more restrictive at higher power levels. It provides a reactor trip if actual ASI exceeds the APD trip setpoint.

Bistable Trip Units

Bistable trip units, mounted in the RPS cabinet, receive an analog input from the measurement channels, compare the analog input to trip setpoints, and provide contact output to the matrix logic. They also provide local trip indication and remote annunciation.

There are four channels of bistable trip units, designated A through D, for each RPS Function, one for each measurement channel. Bistable output relays de-energize when a trip occurs.

The contacts from these bistable relays are arranged into six coincidence matrices, comprising the matrix logic. If bistables monitoring the same parameter in at least two bistable trip unit channels trip, the matrix logic will generate a reactor trip (two-out-of-four logic).

Some of the RPS measurement channels provide contact outputs to the RPS, so the comparison of an analog input to a trip setpoint is not necessary. In these cases, the bistable trip unit is replaced with an auxiliary trip unit. The auxiliary trip units provide contact multiplication so the

single input contact opening can provide multiple contact outputs to the matrix logic, as well as trip indication and annunciation.

Trip Functions employing auxiliary trip units include the Loss of Load trip and the APD trip.

The APD trip, described above, is a complex function in which the actual trip comparison is performed within the APD calculator. Therefore the APD trip unit employs a contact input from the APD calculator.

All RPS trips, with the exception of the Loss of Load trip, generate a pretrip alarm as the trip setpoint is approached.

The trip setpoints used in the bistable trip units are based on the analytical limits stated in Reference 1, Chapter 14, except for the APD and Loss of Load Functions, which are not credited in safety analyses. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account in the respective analytical limits. To allow for calibration tolerances, instrumentation uncertainties, instrument channel drift, and severe environment errors (for those RPS channels that must function in harsh environments, as defined by Reference 2, 10 CFR 50.49) RPS trip setpoints are conservatively adjusted with respect to the analytical limits. In the case of the TM/LP trip, there is also an additional adjustment for cold leg temperature differences. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in Reference 4. The nominal trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value. A channel is inoperable if its actual setpoint is not within its required Allowable Value.

Setpoints in accordance with the Allowable Value will ensure that SLs of Chapter 2.0 are not violated during AOOs and the consequences of DBAs will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed.

Note that in the accompanying LCO 3.3.1, the Allowable Values of Table 3.3.1-1 are the LSSS.

RPS Logic

The RPS logic, addressed in LCO 3.3.3, consists of both matrix and trip path logic and employs a scheme that provides a reactor trip when bistables in any two of the four channels sense the same input parameter trip signal. This is called a two-out-of-four trip logic. This logic and the RTCB configuration are shown in Figure B 3.3.1-1.

Bistable relay contact outputs from the four bistable trip unit channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two bistable trip unit channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable trip unit channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected, consisting of a trip in the same Function in the two channels being monitored by the logic matrix, all four matrix relays de-energize.

The logic matrix relay contacts are arranged into trip paths, with one of the four matrix relays in each matrix opening contacts in one of the four trip paths. Each trip path provides power to one of the four normally energized RTCB control relays (K1, K2, K3, and K4). Thus, the trip paths each have six contacts in series, one from each matrix, performing a logical OR function by opening the RTCBs if any one or more of the six logic matrices indicate a coincidence condition.

Each trip path is responsible for opening one set of two of the eight RTCBs. When de-energized, the RTCB control relays (K-relays) interrupt power to the breaker undervoltage trip coils and simultaneously apply power to the shunt trip coils on each of the two breakers. Actuation of either the undervoltage or shunt trip coil is sufficient to open the RTCB and interrupt power from the motor generator (MG) sets to the control element drive mechanisms (CEDMs).

BASES

When a coincidence occurs in two RPS instrument channels from one Function, all four matrix relays in the affected matrix de-energize. This, in turn, de-energizes all four RTCB control relays, which simultaneously de-energize the undervoltage and energize the shunt trip coils in all eight RTCBs, tripping them open.

Matrix logic refers to the matrix power supplies, trip channel bypass contacts, and interconnecting matrix wiring between bistable and auxiliary trip units, up to but not including the matrix relays. Contacts in the bistable and auxiliary trip units are excluded from the matrix logic definition, since they are addressed separately.

The trip path logic consists of the trip path power source, matrix relays and their associated contacts, and all interconnecting wiring through the K-relay contacts in the RTCB control circuitry.

It is possible to change the two-out-of-four RPS logic to a two-out-of-three logic for a given input parameter, in one channel at a time, by trip bypassing select portions of the matrix logic. Trip bypassing a bistable trip unit effectively shorts the bistable relay contacts in the three matrices associated with that instrument channel. Thus, the bistables will function normally, producing normal trip indication and annunciation, but a reactor trip will not occur unless two additional instrument channels indicate a trip condition. Trip bypassing can be simultaneously performed on any number of parameters in any number of Functions, providing each parameter is bypassed in only one instrument channel per function at a time. Administrative controls prevent simultaneous trip bypassing of the same parameter in more than one instrument channel. Trip bypassing is normally employed during maintenance or testing.

In addition to the trip bypasses, there are also operating bypasses on select RPS trips. Some of these operating bypasses are enabled manually, others automatically, in all four RPS instrument channels for a Function when plant conditions do not warrant the specific trip function protection. All operating bypasses are automatically

removed when enabling bypass conditions are no longer satisfied. Trip Functions with operating bypasses include Rate of Change of Power-High, Reactor Coolant Flow-Low, Steam Generator Pressure-Low, APD-High, TM/LP, and Steam Generator Pressure Difference trips. The Loss-of-Load, Rate of Change of Power-High, and APD-High trips' operating bypasses are automatically enabled and disabled.

RTCBs

The reactor trip switchgear, addressed in LCO 3.3.3 and shown in Figure B 3.3.1-1, consists of eight RTCBs, which are operated in four sets of two breakers (four RTCB channels, including shunt trip coils and undervoltage coils). Power input to the reactor trip switchgear comes from two full capacity MG sets operated in parallel such that the loss of either MG set does not de-energize the CEDMs. There are two separate CEDM power supply buses, each bus powering half of the CEDMs. Power is supplied from the MG sets to each bus via two redundant trip paths. This ensures that a fault or the opening of a breaker in one trip path (i.e., for testing purposes) will not interrupt power to the CEDM buses.

Each of the four trip paths consists of two RTCBs in series. The two RTCBs within a trip path are actuated by separate trip paths.

The eight RTCBs are operated as four sets of two breakers (four RTCB channels, including shunt trip coils and undervoltage coils). Each set of two RTCBs is opened by the same K-relay. This arrangement ensures that power is interrupted to both CEDM buses, thus preventing trip of only half of the CEAs (a half trip). Any one inoperable RTCB in a RTCB channel (set of two breakers) will make the entire RTCB channel inoperable.

Each set of RTCBs is operated by either a manual trip push button or an RPS actuated K-relay. There are four manual trip push buttons, arranged in two sets of two, as shown in Figure B 3.3.1-1. Depressing both push buttons in either set will result in a reactor trip.

BASES

When a manual trip is initiated using the control room push buttons, the RPS trip paths and K-relays are bypassed, and the RTCB undervoltage and shunt trip coils are actuated independent of the RPS.

A manual trip channel includes the push button and interconnecting wiring to both RTCBs necessary to actuate both the undervoltage and shunt trip coils but excludes the K-relay contacts and their interconnecting wiring to the RTCBs, which are considered part of the trip path logic.

Functional testing of the RPS instrument and logic channels, from bistable input through the opening of individual sets of RTCBs, can be performed either at power or shutdown and is normally performed on a quarterly basis. Reference 1, Section 7.2 explains RPS testing in more detail.

APPLICABLE SAFETY ANALYSES

Most of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in Reference 1, Chapter 14 takes credit for most RPS trip Functions. Some Functions not specifically credited in the accident analysis are part of the Nuclear Regulatory Commission (NRC)-approved licensing basis for the plant. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. Other Functions, such as the Loss of Load trip, are purely equipment protective, and their use minimizes the potential for equipment damage.

The specific safety analyses applicable to each protective Function are identified below:

1. <u>Power Level-High Trip</u>

   The Power Level-High trip provides reactor core protection against positive reactivity excursions that are too rapid for a Pressurizer Pressure-High or TM/LP trip to protect against. The following events require Power Level-High trip protection:

   - Uncontrolled CEA withdrawal event;
   - Excess load; and
   - CEA ejection event.

The first two events are AOOs, and fuel integrity is maintained. The third is an accident, and limited fuel damage may occur.

2. Rate of Change of Power-High Trip

The Rate of Change of Power-High trip is used to trip the reactor when excore logarithmic power, measured by the wide range logarithmic neutron flux monitors, indicates an excessive rate of change. The Rate of Change of Power-High trip Function minimizes transients for events such as a born dilution event, continuous CEA withdrawal, or CEA ejection from subcritical conditions. Because of this Function, such events are assured of having much less severe consequences than events initiated from critical conditions. The trip is automatically bypassed when NUCLEAR INSTRUMENT POWER is < 1E-4% RTP, when poor counting statistics may lead to erroneous indication. It is also bypassed at > 12% RTP, where other RPS trips provide protection from these events. With the RTCBs open, the Rate of Change of Power-High trip is not required to be OPERABLE; however, at least two wide range logarithmic neutron flux monitor channels are required by LCO 3.3.12 to be OPERABLE. Limiting Condition for Operation 3.3.12 ensures the wide range logarithmic neutron flux monitor channels are available to detect and alert the operator to a boron dilution event.

3. Reactor Coolant Flow-Low Trip

The Reactor Coolant Flow-Low trip provides protection during the following events:

- Loss of RCS flow;
- Loss of non-emergency AC power; and
- Reactor coolant pump (RCP) seized rotor.

The loss of RCS flow and of non-emergency AC power events are AOOs where fuel integrity is maintained. The RCP seized rotor is an accident where fuel damage may result.

4. <u>Pressurizer Pressure-High Trip</u>

The Pressurizer Pressure-High trip, in conjunction with pressurizer safety valves and main steam safety valves, provides protection against overpressure conditions in the RCS during the following events:

- Loss of Load; and
- Feedwater Line Break (FWLB).

5. <u>Containment Pressure-High Trip</u>

The Containment Pressure-High trip prevents exceeding the containment design pressure during certain loss of coolant accidents (LOCAs) or FWLB accidents. It ensures a reactor trip prior to, or concurrent with, a LOCA, thus assisting the ESFAS in the event of a LOCA or Main Steam Line Break (MSLB). Since these are accidents, SLs may be violated. However, the consequences of the accident will be acceptable.

6. <u>Steam Generator Pressure-Low Trip</u>

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators, which would result in a rapid uncontrolled cooldown of the RCS. This trip is needed to shut down the reactor and assist the ESFAS in the event of an MSLB. Since these are accidents, SLs may be violated. However, the consequences of the accident will be acceptable.

7. <u>Steam Generator 1 and 2 Level-Low Trip</u>

The Steam Generator 1 Level-Low and Steam Generator 2 Level-Low trips are required for the loss of normal feedwater and ASGT events.

The Steam Generator Level-Low trip ensures that low DNBR, high local power density, and the RCS pressure SLs are maintained during normal operation and AOOs, and, in conjunction with the ESFAS, the consequences of the Feedwater System pipe break accident will be acceptable.

8. APD-High Trip

The APD-High trip ensures that excessive axial peaking, such as that due to axial xenon oscillations, will not cause fuel damage. It ensures that neither a DNBR less than the SL, nor a peak linear heat rate that corresponds to the temperature for fuel centerline melting, will occur. This trip is the primary protection against fuel centerline melting. While no event specifically credits the Axial Flux Offset trip, the ASI limits established by this trip provide ASI limits for safety and setpoint analyses.

9. Thermal Margin

a. TM/LP Trip

The TM/LP trip prevents exceeding the DNBR SL during AOOs and aids the ESFAS during certain accidents. The following events require TM/LP trip protection:

- RCS depressurization (inadvertent safety or power-operated relief valves opening);
- Steam generator tube rupture; and
- LOCA accident.

The first event is an AOOs, and fuel integrity is maintained. The second and third events are accidents, and limited fuel damage may occur, although only the LOCA is expected to result in fuel damage. The trip is initiated whenever the RCS pressure signal drops below a minimum value (P_{min}) or a computed value (P_{var}) as described below, whichever is higher. The setpoint is a Function of Q power, ASI, and reactor inlet (cold leg) temperature.

The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT (T_q), and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip signal. In addition, CEA group sequencing in accordance with LCO 3.1.7 is assumed. Finally,

the maximum insertion of CEA banks that can occur during any AOO prior to a Power Level-High trip is assumed.

b. ASGT

The ASGT provides protection for those AOOs associated with secondary system malfunctions that result in asymmetric primary coolant temperatures. The most limiting event is closure of a single main steam isolation valve (MSIV). Asymmetric Steam Generator Transient is provided by comparing the secondary pressure in both steam generators in the TM/LP trip calculator. If the pressure in either exceeds that in the other by the trip setpoint, a TM/LP trip will result.

10. Loss of Load

The Loss of Load trip causes a trip when operating above 15% of RTP. This trip provides turbine protection, reduces the severity of the ensuing transient, and helps avoid the lifting of the main steam safety valves during the ensuing transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability is required to enhance overall plant equipment service life and reliability.

Operating Bypasses

The operating bypasses are addressed in footnotes to Table 3.3.1-1. They are not otherwise addressed as specific table entries.

The automatic bypass removal features must function as a backup to manual actions for all trips credited in safety analyses to ensure the trip Functions are not operationally bypassed when the safety analysis assumes the Functions are not bypassed. The RPS operating bypasses are:

Zero power mode bypass (ZPMB) removal of the TM/LP, ASGT, and reactor coolant low flow trips when NUCLEAR INSTRUMENT POWER is < 1E-4% RTP. This bypass is manually enabled below

difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.2) and the monthly linear subchannel gain check (SR 3.3.1.3).

SR 3.3.1.9

This SR ensures that the RPS RESPONSE TIMES are verified to be less than or equal to the maximum values assumed in the safety analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the RTCBs open. Response times are conducted on a 24-month STAGGERED TEST BASIS. Response time testing acceptance criteria are included in Reference 1, Section 7.2. This results in the interval between successive SRs of a given channel of n x 24 months, where n is the number of channels in the function. The Frequency of 24 months is based upon operating experience, which has shown that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. Also, response times cannot be determined at power since equipment operation is required. Testing may be performed in one measurement or in overlapping segments, with verification that all components are tested.

Response time may be verified by any series of sequential, overlapping or total channel measurements, including allocated sensor response time, such that the response time is verified. Allocations for sensor response times may be obtained from records of test results, vendor test data, or vendor engineering specifications. Reference 9 provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the reference. Response time verification for other sensor types must be demonstrated by test. The allocation of sensor response times must be verified prior to placing a new component in operation and reverified after maintenance that may adversely affect the sensor response time.

BASES

Instrument loop or test cables and wiring add an insignificant response time and can be ignored.

A Note is added to indicate that the neutron detectors are excluded from RPS RESPONSE TIME testing because they are passive devices with minimum drift, and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.3).

REFERENCES

1. Updated Final Safety Analysis Report
2. Title 10 Code of Federal Regulations
3. Institute of Electrical and Electronic Engineers (IEEE) No. 279, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems," August 1968
4. CCNPP Setpoint File
5. Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated June 5, 1995, "Response to NRC Request for Review & Comment on Review of Preliminary Accident Precursor Analysis of Trip; Loss of 13.8 kV Bus; Short-Term Saltwater Cooling System Unavailability, CCNPP Unit 2"
6. Combustion Engineering Topical Report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" dated June 2, 1986, including Supplement 1, March 3, 1989
7. Letter from Mr. D. G. McDonald (NRC) to Mr. R. E. Denton (BGE), dated October 19, 1995, "Issuance of Amendments for Calvert Cliffs Nuclear Power Plant, Unit No. 1 (TAC No. M92479) and Unit No. 2 (TAC No. M92480)
8. Calvert Cliffs Procedure EN-4-104, "Surveillance Testing"
9. Combustion Engineering Owners Group Topical Report CE NPSD 1167-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements", July 3, 2000

every 24 months and is only applicable to automatic block removal features of the sensor block modules. These include the Pressurizer Pressure-Low trip block and the SGIS Steam Generator Pressure-Low trip block.

The CHANNEL FUNCTIONAL TEST for proper operation of the automatic block removal features is critical during plant heatups because the blocks may be in place prior to entering MODE 3, but must be removed at the appropriate points during plant startup to enable the ESFAS Function. A 24-month SR Frequency is adequate to ensure proper automatic block removal module operation as described in Reference 3. Once the blocks are removed, the blocks must not fail in such a way that the associated ESFAS Function is inappropriately blocked. This feature is verified by the appropriate ESFAS Function CHANNEL FUNCTIONAL TEST.

The 24-month SR Frequency is adequate to ensure proper automatic block removal feature operation as described in Reference 3.

SR 3.3.4.4

CHANNEL CALIBRATION is a check of the sensor channel, including the automatic block removal feature of the sensor block module and the sensor. The SR verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for sensor channel drift between successive calibrations to ensure that the channel remains operational between successive surveillance tests. CHANNEL CALIBRATIONS must be performed consistent with Reference 5.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the extension analysis. The requirements for this review are outlined in Reference 6.

The Frequency is based upon the assumption of a 24-month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.4.5

This SR ensures that the train actuation response times are the maximum values assumed in the safety analyses. Individual component response times are not modeled in the analyses. The analysis models the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment in both trains reaches the required functional state (e.g., pumps are rated discharge pressure, valves in full open or closed position). Response time testing acceptance criteria are included in Reference 1, Section 7.3. The test may be performed in one measurement or in overlapping segments, which verification that all components are measured.

Response time may be verified by any series of sequential, overlapping or total channel measurements, including allocated sensor response time, such that the response time is verified. Allocations for sensor response times may be obtained from records of test results, vendor test data, or vendor engineering specifications. Reference 7 provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the reference. Response time verification for other sensor types must be demonstrated by test. The allocation of sensor response times must be verified prior to placing a new component in operation and reverified after maintenance that may adversely affect the sensor response time.

Instrument loop or test cables and wiring add an insignificant response time and can be ignored.

Engineered Safety Feature Response Time tests are conducted on a STAGGERED TEST BASIS of once every 24 months. This results in the interval between successive tests of a given channel of n x 24 months, where n is the number of channels in the Function. Surveillance of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. Therefore, staggered testing results in Response Time verification of these devices every 24 months. The 24-month STAGGERED TEST

BASES

BASIS Frequency is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

REFERENCES

1. UFSAR
2. IEEE No. 279, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems," August 1968
3. Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated June 5, 1995, "Response to NRC Request for Review & Comment on Review of Preliminary Accident Precursor Analysis of Trip; Loss of 13.8 kV Bus; Short-Term Saltwater Cooling System Unavailability, CCNPP Unit 2"
4. 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"
5. CCNPP Setpoint File
6. Calvert Cliffs Procedure EN-4-104, "Surveillance Testing"
7. Combustion Engineering Owners Group Topical Report CE NPSD 1167-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements", July 3, 2000

explicitly account for each individual component of the loss of power detection and subsequent actions. This delay time includes contributions from the DG start, DG loading, and Safety Injection System component actuation. The response of the DG to a loss of power must be demonstrated to fall within this analysis response time when including the contributions of all portions of the delay.

The required channels of LOVS, in conjunction with the ESF systems powered from the DGs, provide plant protection in the event of any of the analyzed accidents discussed in Reference 1, Chapter 8 in which a loss of offsite power is assumed. Loss of voltage start channels are required to meet the redundancy and testability requirements of Reference 1, Appendix 1C.

The delay times assumed in the safety analysis for the ESF equipment include the 10-second DG start delay and the appropriate sequencing delay, if applicable. The response times for ESFAS-actuated equipment include the appropriate DG loading and sequencing delay.

The DG-LOVS channels satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The LCO for the LOVS requires that four channels per bus of each LOVS instrumentation Function be OPERABLE in MODEs 1, 2, 3, and 4. The LOVS supports safety systems associated with the ESFAS.

Actions allow maintenance bypass of individual sensor channels. The plant is restricted to 48 hours in a maintenance bypass condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic).

Loss of LOVS Function could result in the delay of safety system actuation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite power, which is a AOO, the DG powers the motor-driven AFW pump. Failure of this pump to start would leave two turbine-driven pumps as well as an increased potential

for a loss of decay heat removal through the secondary system.

Only Allowable Values are specified for each Function in the LCO. Nominal trip settings are specified in the plant-specific procedures. The nominal settings are selected to ensure that the setting measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setting less conservative than the nominal trip setting, but within the Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the plant-specific setting calculation. A channel is inoperable if its actual trip setting is not within its required Allowable Value.

The Allowable Values and trip settings are established in order to start the DGs at the appropriate time, in response to plant conditions, in order to provide emergency power to start and supply the essential electrical loads necessary to safely shut down the plant and maintain it in a safe shutdown condition.

APPLICABILITY

The DG-LOVS actuation Function is required in MODEs 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODEs.

ACTIONS

A LOVS sensor channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's Function. The most common cause of sensor channel inoperability is outright failure or drift of the bistable (sensor module) or measurement channel sufficient to exceed the tolerance allowed by the plant-specific setting analysis. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. Determination of setting drift is generally made during the performance of a CHANNEL CALIBRATION when the process instrument is set up for adjustment to bring it to within specification. CHANNEL FUNCTIONAL TESTS identify sensor module drift. If the actual trip setting is not within the Allowable Value, the channel is inoperable and the appropriate Conditions must be entered.

In the event a sensor channel's setting is found to be nonconservative with respect to the Allowable Value, or the channel is found to be inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition entered. The required channels are specified on a per DG basis.

When the number of inoperable channels in a Function exceeds those specified in any related Condition associated with the same Function, the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be entered immediately if applicable in the current MODE of operation.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this LCO may be entered independently for each Function. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function, starting from the time the Condition was entered for that Function.

A.1, A.2.1, and A.2.2

Condition A applies if one sensor channel is inoperable for one or more Functions per DG bus.

If the channel cannot be restored to OPERABLE status, the affected channel should either be bypassed or tripped within 1 hour (Required Action A.1).

Placing this channel in either Condition ensures that logic is in a known configuration. In trip, the LOVS logic is one-out-of-three. In bypass, the LOVS logic is two-out-of-three. The 1-hour Completion Time is sufficient to perform these Required Actions.

Once Required Action A.1 has been complied with, Required Action A.2.1 allows 48 hours to repair the inoperable sensor channel. If the channel cannot be restored to OPERABLE status, it must be tripped in accordance with Required Action A.2.2. The time allowed to repair or trip the channel is reasonable to repair the affected channel while ensuring that the risk involved in operating with the

inoperable channel is acceptable. The 48-hour Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel is a rare event during any given 48-hour period.

B.1, B.2.1, and B.2.2

Condition B applies if two sensor channels are inoperable for one or more Functions per DG.

The Required Action is modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODEs, even though two sensor channels are inoperable, with one channel bypassed and one tripped. In this configuration, the protective system is in a one-out-of-two logic, which is adequate to ensure that no random failure will prevent protective system operation.

Restoring at least one channel to OPERABLE status is the preferred action. If the channel cannot be restored to OPERABLE status within 1 hour, the Conditions and Required Actions for the associated DG made inoperable by DG-LOVS instrumentation are required to be entered. Alternatively, one affected channel is required to be bypassed and the other is tripped, in accordance with Required Action B.2.1. This places the Function in one-out-of-two logic. The 1-hour Completion Time is sufficient to perform the Required Actions.

Once Required Action B.2.1 has been complied with, Required Action B.2.2 allows 48 hours to repair the bypassed or inoperable channel.

After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2.2 shall be placed in trip if more than 48 hours have elapsed since the initial channel failure.

C.1

Condition C applies when more than two undervoltage or degraded (transient or steady state) voltage sensor channels on a single bus are inoperable.

Required Action C.1 requires all but two channels to be restored to OPERABLE status within 1 hour. With more than two channels inoperable, the logic is not capable of providing a DG-LOVS signal for valid loss of voltage or degraded voltage conditions. The 1 hour Completion Time is reasonable to evaluate and take action to correct the degraded condition in an orderly manner and takes into account the low probability of an event requiring LOVS occurring during this interval.

D.1

Condition D applies if the Required Actions and associated Completion Times are not met.

Required Action D.1 ensures that Required Actions for the affected DG inoperabilities are initiated. The actions specified in LCO 3.8.1 are required immediately.

SURVEILLANCE REQUIREMENTS

The following SRs apply to each DG-LOVS Function.

SR 3.3.6.1

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure that the entire sensor channel will perform its intended function when needed.

The Frequency of 92 days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one sensor channel of a given function in any 92 day Frequency is a rare event. Any setting adjustment shall be consistent with the assumptions of the current plant specific setting analysis.

SR 3.3.6.2

Surveillance Requirement 3.3.6.2 is the performance of a CHANNEL CALIBRATION every 24 months. The CHANNEL

BASES

CALIBRATION verifies the accuracy of each component within the sensor channel, except stepdown transformers, which are not calibrated. This includes calibration of the undervoltage relays and demonstrates that the equipment falls within the specified operating characteristics defined by the manufacturer.

The SR verifies that the sensor channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the plant-specific setting analysis.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the SR interval extension analysis. The requirements for this review are outlined in Reference 4.

The settings, as well as the response to Loss of Voltage and Degraded Voltage tests, shall include a single point verification that the trip occurs within the required delay time as shown in Reference 1, Section 7.3. The Frequency is based upon the assumption of a 24-month calibration interval for the determination of the magnitude of equipment drift in the plant setting analyses.

REFERENCES

1. UFSAR
2. CCNPP Setpoint File
3. IEEE No. 279, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems," August 1968
4. Calvert Cliffs Procedure EN-4-104, "Surveillance Testing"

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power-Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORV is an electric, solenoid-operated valve that is automatically opened at a specific set pressure when the pressurizer pressure increases and is automatically closed on decreasing pressure. The PORV may also be manually opened or closed using a handswitch installed in the Control Room.

An electric, motor-operated, normally open, block valve is installed between the pressurizer and the PORV. The function of the block valve is to isolate the PORV. Block valve closure is accomplished manually using controls in the Control Room and may be used to isolate a leaking PORV to permit continued power operation. Most importantly, the block valve is used to isolate a stuck open PORV to isolate the resulting small break LOCA. Closure terminates the RCS depressurization and coolant inventory loss.

The PORV and its block valve controls are powered from normal power supplies. Their controls are also capable of being powered from emergency supplies. Power supplies for the PORV are separate from those for the block valve. Power supply requirements are defined in Reference 1.

The PORV setpoint is equal to the high pressure reactor trip setpoint and below the opening setpoint for the pressurizer safety valves as required by Reference 2. The purpose of the relationship of these setpoints is to reduce the frequency of challenges to the safety valves, which, unlike the PORV, cannot be isolated if they were to fail open.

The primary purpose of this LCO is to ensure that the PORV and the block valve are operating correctly so the potential for a small break LOCA through the PORV pathway is minimized; or if a small break LOCA were to occur through a failed open PORV, the block valve could be manually operated to isolate the path.

The PORV may be manually-operated to depressurize the RCS as deemed necessary by the operator in response to normal or abnormal transients. The PORV may be used for depressurization when the pressurizer spray is not available, a condition that may be encountered during loss of offsite power. Operators can manually open the PORVs to reduce RCS pressure in the event of a steam generator tube rupture (SGTR) with offsite power unavailable.

The PORV may also be used for once through core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORV functions as an automatic overpressure device and limits challenges to the safety valves. Although the PORV acts as an overpressure device for operational purposes, safety analyses do not take credit for PORV actuation, but do take credit for the safety valves.

The PORV also provides LTOP during heatup and cooldown. Limiting Condition for Operation 3.4.12, addresses this function.

APPLICABLE SAFETY ANALYSES

The PORV small break LOCA break size is bounded by the spectrum of piping breaks analyzed for plant licensing. Because the PORV small break LOCA is located at the top of the pressurizer, the RCS response characteristics are different from RCS loop piping breaks; analyses have been performed to investigate these characteristics.

The possibility of a small break LOCA through the PORV is reduced when the PORV flow path is OPERABLE. The possibility is minimized if the flow path is isolated.

Overpressure protection is provided by safety valves, and analyses do not take credit for the PORV opening for accident mitigation.

Pressurizer PORVs satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The LCO requires the two PORVs and their associated block valves to be OPERABLE. The block valve is required to be

BASES

OPERABLE so it may be used to isolate the flow path if the PORV is not OPERABLE.

Valve OPERABILITY also means the PORV setpoint is correct. Ensuring the PORV opening setpoint is correct reduces the frequency of challenges to the safety valves, which, unlike the PORVs, cannot be isolated if they were to fail open.

APPLICABILITY

In MODEs 1 and 2, and MODE 3 with all RCS cold leg temperatures > 365°F (Unit 1), > 301°F (Unit 2), the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. A likely cause for PORV small break LOCA is a result of pressure increase transients that cause the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. Pressure increase transients can occur any time the SGs are used for heat removal. The most rapid increases will occur at higher operating power and pressure conditions of MODEs 1 and 2.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, this LCO is applicable in MODEs 1 and 2, and MODE 3 with all RCS cold leg temperatures > 365°F (Unit 1), > 301°F (Unit 2). The LCO is not applicable in MODE 3 with all RCS cold leg temperatures ≤ 365°F (Unit 1), ≤ 301°F (Unit 2), when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODE 3 with T_{avg} ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) and in MODEs 4, 5, and 6 with the reactor vessel head in place. Limiting Condition for Operation 3.4.12 addresses the PORV requirements in these MODEs.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 clarifies that the pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). Note 2 is an exception to LCO 3.0.4. The exception to LCO 3.0.4 permits entry into MODEs 1, 2, and 3 to perform cycling of the PORV or block

valve to verify their OPERABLE status. Testing is typically not performed in lower MODEs.

A.1

With one or two PORVs inoperable and capable of being manually cycled, either the inoperable PORV(s) must be restored or the flow path isolated within one hour. The block valve should be closed but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. Although the PORV may be designated inoperable, it may be able to be manually opened and closed, and in this manner can be used to perform its function. Power-operated relief valve inoperability may be due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use, and do not create a possibility for a small break LOCA. For these reasons, the block valve may be closed but the Action requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. The PORVs should normally be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2).

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of one hour is based on plant operating experience that minor problems can be corrected or closure can be accomplished in this time period.

B.1, B.2, and B.3

If one PORV is inoperable and not capable of being manually cycled, it must either be isolated, by closing the associated block valve and removing the power from the block valve, or restored to OPERABLE status. The Completion Time of one hour is reasonable, based on challenges to the PORVs during this time period, and provides the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one

PORV that remains OPERABLE, five days are provided to restore the inoperable PORV to OPERABLE status.

C.1 and C.2

If one block valve is inoperable, then it must be restored to OPERABLE status, or the associated PORV placed in override closed. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within one hour, the Required Action is to place the PORV in override closed to preclude its automatic opening for an overpressure event, and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Times of one hour are reasonable based on the small potential for challenges to the system during this time period and provide the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of five days to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B since the PORVs are not capable of automatically mitigating an overpressure event when placed in override closed. If the block valve is restored within the Completion Time of five days, the power will be restored and the PORV restored to OPERABLE status.

D.1, D.2, and D.3

If both PORVs are inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of one hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of one hour is reasonable based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If Required Actions D.1 and D.2 have been completed, Required Action D.3 allows 72 hours to restore a PORV to OPERABLE status. This time is reasonable to perform required repairs. This time also accounts for the overpressure protection provided by the pressurizer safety valves in LCO 3.4.10.

E.1 and E.2

If both block valves are inoperable, it is necessary to either restore the block valves within the Completion Time of one hour or place the associated PORVs in override closed and restore at least one block valve to OPERABLE status within 72 hours, and the remaining block valve in five days, per Required Action C.2. The Completion Time of one hour to either restore the block valves or place the associated PORVs in override closed is reasonable based on the small potential for challenges to the system during this time and provides the operator time to correct the situation.

F.1 and F.2

If the Required Actions and associated Completion Times are not met, then the plant must be brought to a MODE in which the LCO does not apply. The plant must be brought to at least MODE 3 within 6 hours and reduce any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) within 12 hours. The Completion Time of six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 12 hours to reduce any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) is reasonable considering that a plant can cool down within that time frame. In MODE 3 with any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) and in MODEs 4, 5, and 6, maintaining PORV OPERABILITY is required per LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

A CHANNEL FUNCTIONAL TEST is performed on each PORV instrument channel every 92 days to ensure the entire channel will perform its intended function when needed.

SR 3.4.11.2

Block valve cycling verifies that it can be closed if necessary. The basis for the Frequency of 92 days is found in Reference 3. If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance because

BASES

opening the block valve is necessary to permit the PORV to be used for manual control of RCS pressure. If the block valve is closed to isolate an otherwise inoperable PORV, the maximum Completion Time to restore the PORV and open the block valve is 120 hours, which is well within the allowable limits (25%) to extend the block valve surveillance interval of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status (i.e., completion of the Required Action fulfills the SR).

The Note modifies this SR by stating that this SR is not required to be performed with the block valve closed in accordance with the Required Actions of this LCO.

SR 3.4.11.3

Surveillance Requirement 3.4.11.3 requires complete cycling of each PORV. Power-operated relief valve cycling demonstrates its function. The Frequency of 24 months is based on a typical refueling cycle and industry accepted practice.

SR 3.4.11.4

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 24 months to adjust the whole channel so that it responds, and the valve opens within the required range and with accuracy to known input.

The 24 month Frequency considers operating experience with equipment reliability and matches the refueling outage Frequency.

REFERENCES

1. NUREG-0737, Paragraph II, G.I, "Clarification of TMI Action Plan Requirements," November 1980
2. Inspection and Enforcement.Bulletin 79-05B, "Nuclear Incident at Three Mile Island - Supplement," April 21, 1979
3. ASME, Boiler and Pressure Vessel Code, Section XI

BASES

Compliance with this LCO will ensure a containment configuration, including an equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) are not specifically part of the acceptance criteria of Reference 1. Therefore, leakage rates exceeding these individual limits only result in the Containment Structure being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

APPLICABILITY

In MODEs 1, 2, 3, and 4, a DBA could cause a release of radioactive material into the Containment Structure. In MODEs 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODEs. Therefore, the Containment Structure is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from the Containment Structure. The requirements for the Containment Structure during MODE 6 are addressed in LCO 3.9.3.

ACTIONS

A.1

In the event the Containment Structure is inoperable, the Containment Structure must be restored to OPERABLE status within one hour. The one hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining the Containment Structure during MODEs 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring Containment OPERABILITY) occurring during periods when the Containment Structure is inoperable is minimal.

B.1 and B.2

If the Containment Structure cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the Containment Structure OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage, prior to the first startup after performing a required Containment Leakage Rate Testing Program, is required to be $\leq 0.6\ L_a$ (207,600 SCCM) for combined Type B and C leakage and $\leq 0.75\ L_a$ (259,500 SCCM) for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$, the offsite dose consequences are bounded by the assumptions of the safety analysis. Surveillance Requirement Frequencies are as required by Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

For ungrouted, post-tensioned tendons, this SR ensures that the structural integrity of the Containment Structure will be maintained in accordance with the provisions of the Concrete Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Reference 3.

BASES

This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences of accidents that could result in a challenge to the reactor coolant pressure boundary.

APPLICABILITY

In MODEs 1, 2, and 3, a minimum of five MSSVs per steam generator are required to be OPERABLE, according to Table 3.7.1-1 in the accompanying LCO, which is limiting and bounds all lower MODEs.

In MODEs 4 and 5, there are no credible transients requiring the MSSVs.

The steam generators are not normally used for heat removal in MODEs 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODEs.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets Code requirements for the power level. The number of inoperable MSSVs will determine the necessary level of reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the power level-high channels. The reactor trip setpoint reductions are derived on the following basis:

$$SP = \frac{(X) - (Y)(V)}{X} \times 106.5$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

106.5 = Power Level-High Trip Setpoint

X = Total relieving capacity of all safety valves per steam line in lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour

Nuclear Regulatory Commission Information Notice 94-60 states that the linear relationship is not always valid; however, the setpoints in Table 3.7.1-1 have been verified by transient analyses.

The operator should limit the maximum steady state power level to some value slightly below this setpoint to avoid an inadvertent overpower trip.

The four-hour Completion Time for Required Action A.1 is a reasonable time period to reduce power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs. An additional eight hours is allowed in Required Action A.2 to reduce the setpoints. The Completion Time of 12 hours for Required Action A.2 is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status in the associated Completion Time, or if one or more steam generators have less than five MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This Surveillance Requirement (SR) verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoints in accordance with the Inservice Testing Program.

BASES

Reference 2, Section XI, Article IWV-3500, requires that safety and relief valve tests be performed in accordance with Reference 3. According to Reference 3, the following tests are required for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

The ANSI/American Society of Mechanical Engineers (ASME) Standard requires that all valves be tested every five years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities, as found lift acceptance range, and frequencies necessary to satisfy the requirements. Table 3.7.1-2 defines the lift setting range for each MSSV for OPERABILITY; however, the valves are reset to $\pm$ 1% during the surveillance test to allow for drift.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This is to allow testing of the MSSVs at hot conditions. The MSSVs may be either bench tested or tested in situ at hot conditions, using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. Updated Final Safety Analysis Report (UFSAR)
2. ASME, Boiler and Pressure Vessel Code
3. ANSI/ASME OM-1-1987, Code for the Operation and Maintenance of Nuclear Power Plants, 1987

BASES

The AFW System satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

This LCO requires that two AFW trains be OPERABLE to ensure that the AFW System will perform its design safety function. A train consists of one pump and the piping, valves, and controls in the direct flow path. Three AFW pumps are installed, consisting of one motor-driven and two non-condensing steam turbine-driven pumps. For a shutdown, only one pump is required to be operating, the others are in standby. Upon automatic initiation of AFW, one motor-driven and one turbine-driven pump automatically start.

The AFW System is considered to be OPERABLE when the components and flow paths required to provide AFW flow to the steam generators are OPERABLE. This requires that the motor-driven AFW pump be OPERABLE and capable of supplying AFW flow to both steam generators. The turbine-driven AFW pumps shall be OPERABLE with redundant steam supplies from each of the two main steam lines upstream of the MSIVs and capable of supplying AFW flow to both of the two steam generators. The piping, valves, instrumentation, and controls in the required flow paths shall also be OPERABLE.

The LCO is modified by a Note that allows AFW trains required for Operability to be taken out-of-service under administrative control for the performance of periodic testing. This LCO note allows a limited exception to the LCO requirement and allows this condition to exist without requiring any Technical Specification Condition to be entered. The following administrative controls are necessary during periodic testing to ensure the operator(s) can restore the AFW train(s) from the test configuration to its operational configuration when required. A dedicated operator(s) is stationed at the control station(s) with direct communication to the Control Room whenever the train(s) is in the testing configuration. Upon completion of the testing the trains are returned to proper status and verified in proper status by independent operator checks. The administrative controls include certain operator restoration actions that are virtually certain to be successful during accident conditions. These actions

include but are not limited to the following: operation of pump discharge valves, operation of trip/throttle valve(s), simple handswitch/controller manipulations, and adjusting the local governor speed control knob. The administrative controls do not include actions to restore a tripped AFW pump due to the complicated nature of this task. Periodic tests include those tests that are performed in a controlled manner similar to surveillance tests, but not necessarily on the established surveillance test schedule, such as post-maintenance tests. This Note is necessary because of the AFW pump configuration.

APPLICABILITY

In MODEs 1, 2, and 3, the AFW System is required to be OPERABLE and to function in the event that the MFW is lost. In addition, the AFW System is required to supply enough makeup water to replace steam generator secondary inventory and maintain the RCS in MODE 3.

In MODE 4, the AFW System is not required, however, it may be used for heat removal via the steam generator although the preferred method is MFW.

In MODEs 5 and 6, the steam generators are not normally used for decay heat removal, and the AFW System is not required.

ACTIONS

A.1 and A.2

With one of the required steam-driven AFW pumps inoperable, action must be taken to align the remaining OPERABLE steam-driven pump to automatic initiating status. This Required Action ensures that a steam-driven AFW pump is available to automatically start, if required. If the OPERABLE AFW pump is properly aligned, the inoperable steam-driven AFW pump must be restored to OPERABLE status (and placed in either standby or automatic initiating status, depending upon whether the other steam-driven AFW pump is in standby or automatic initiating status) within seven days. The 72 hour and seven day Completion Times are reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. Two AFW pumps and flow paths remain to supply feedwater to the steam generators. The second Completion Time for Required Action A.2 establishes a

limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. The AND connector between seven days and ten days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1 and B.2

With the motor-driven AFW pump inoperable, action must be taken to align the standby steam-driven pump to automatic initiating status. This Required Action ensures that another AFW pump is available to automatically start, if required. If the standby steam-driven pump is properly aligned, the inoperable motor-driven AFW pump must be restored to OPERABLE status within seven days. The 72-hour and seven day, Completion Times are reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. Two AFW pumps and one flow path remain to supply feedwater to the steam generators. The second Completion Time for Required Action B.2 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. The AND connector between seven days and ten days dictates that both Completion Times apply simultaneously, and more restrictive must be met.

C.1, C.2, C.3, and C.4

With two AFW pumps inoperable, action must be taken to align the remaining OPERABLE pump to automatic initiating status and to verify the other units motor-driven AFW pump is OPERABLE, along with an OPERABLE cross-tie valve, within one hour. If these Required Actions are completed within the Completion Time, one AFW pump must be restored to OPERABLE status within 72 hours. Verifying the other unit's

motor-driven AFW pump is OPERABLE provides an additional level of assurance that AFW will be available if needed, because the other unit's AFW can be cross-connected if necessary. The cross-tie valve to the opposite unit is administratively verified OPERABLE by confirming that SR 3.7.3.2 has been performed within the specified Frequency. These one hour Completion Times are reasonable based on the low probability of a DBA occurring during the first hour and the need for AFW during the first hour. The 72 hour completion time to restore one AFW pump to OPERABLE status takes into account the cross-connected capability between units and the unlikelihood of an event occurring in the 72 hour period.

D.1

With one of the required AFW trains inoperable for reasons other than Condition A, B, or C (e.g., flowpath or steam supply valve), action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine-driven AFW pumps. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. One AFW train remains to supply feedwater to the steam generators. The second Completion Time for Required Action D.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The ten day Completion Time provides a limited time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and ten days dictates that both Completion Times apply simultaneously, and more restrictive must be met.

E.1 and E.2

When the Required Action and associated Completion Time of Condition A, B, C, or D cannot be met the unit must be placed in a MODE in which the LCO does not apply. To

achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

F.1

Required Action F.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status.

With two AFW trains inoperable in MODEs 1, 2, and 3, the unit may be in a seriously degraded condition with only non-safety-related means for conducting a cooldown. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. However, a power change is not precluded if it is determined to be the most prudent action. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status. While other plant conditions may require entry into LCO 3.0.3, the ACTIONS required by LCO 3.0.3 do not have to be completed because they could force the unit into a less safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the AFW water and steam supply flow paths, provides assurance that the proper flow paths exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulations; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.3.2

Cycling each testable, remote-operated valve that is not in its operating position, provides assurance that the valves will perform as required. Operating position is the position that the valve is in during normal plant operation. This is accomplished by cycling each valve at least one cycle. This SR ensures that valves required to function during certain scenarios, will be capable of being properly positioned. The Frequency is based on engineering judgment that when cycled in accordance with the Inservice Testing Program, these valves can be placed in the desired position when required.

SR 3.7.3.3

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ($\geq$ 2800 ft for the steam-driven pump and $\geq$ 3100 ft for the motor-driven pump), ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of pump performance required by Reference 2. Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing, discussed in Reference 2, at three month intervals satisfies this requirement.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is an insufficient steam pressure to perform the test.

SR 3.7.3.4

This SR ensures that AFW can be delivered to the appropriate steam generator, in the event of any accident or transient that generates an AFAS signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal (verification of flow-modulating characteristics is not required). This SR is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. The 24 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions have been established.

SR 3.7.3.5

This SR ensures that the AFW pumps will start in the event of any accident or transient that generates an AFAS signal by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. The 24 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.

This SR is modified by a Note. The Note indicates that the SR be deferred until suitable test conditions are established.

SR 3.7.3.6

This SR ensures that the AFW system is capable of providing a minimum nominal flow to each flow leg. This ensures that the minimum required flow is capable of feeding each flow leg. The test may be performed on one flow leg at a time. The SR is modified by a Note which states, the SR is not required to be performed for the AFW train with the turbine-driven AFW pump until 24 hours after reaching 800 psig in the steam generators. The Note ensures that proper test

conditions exist prior to performing the test using the turbine-driven AFW pumps. The 24 month Frequency coincides with performing the test during refueling outages.

SR 3.7.3.7

This SR ensures that the AFW System is properly aligned by verifying the flow path to each steam generator prior to entering MODE 2 operation, after 30 days in MODEs 5 or 6. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, and other administrative controls to ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, the OPERABILITY of the flow paths is verified following extended outages to determine that no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned. Minimum nominal flow to each flow leg is ensured by performance of SR 3.7.3.6.

REFERENCES

1. UFSAR, Section 10.3
2. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400

B 3.7 PLANT SYSTEMS

B 3.7.7 Saltwater (SW) System

BASES

BACKGROUND

The SW System provides a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient. During normal operation or a normal shutdown, the SW System also provides this function for various safety-related and non-safety-related components. The safety-related function is covered by this LCO.

The SW System consists of two subsystems. Each subsystem contains one pump. A third pump, which is an installed spare, can be aligned to either subsystem. The safety-related function of each subsystem is to provide SW to two SRW heat exchangers, a CC heat exchanger, and an Emergency Core Cooling System (ECCS) pump room air cooler in order to transfer heat from these systems to the bay. Seal water for the non-safety-related circulating water pumps is supplied by both or either subsystems. The SW pumps provide the driving head to move SW from the intake structure, through the system and back to the circulating water discharge conduits. The system is designed such that each pump has sufficient head and capacity to provide cooling water such that 100% of the required heat load can be removed by either subsystem.

During normal operation, both subsystems in each unit are in operation with one pump running on each header and a third pump in standby. If needed, the standby pumps can be lined-up to either supply header. The SW flow through the SRW and CC heat exchangers is throttled to provide sufficient cooling to the heat exchangers, while maintaining total subsystem flow below a maximum value.

Additional information about the design and operation of the SW System, along with a list of the components served, is presented in Reference 1. During an accident, the SW System is required to remove the heat load from the SRW and ECCS pump room, and from the CC following an RAS.

BASES

APPLICABLE SAFETY ANALYSES

The most limiting event for the SW System is a LOCA. Operation of the SW System following a LOCA is separated into two phases, before the RAS and after the RAS. One subsystem can satisfy cooling requirements of both phases. After a LOCA but before an RAS, each subsystem will cool two SRW heat exchangers and an ECCS pump room air cooler (as required). There is no required flow to the CC heat exchangers. When an RAS occurs, flow is throttled to the CC heat exchanger. Flow to each SRW heat exchanger is reduced while the system remains capable of providing the required flow to the ECCS pump room air coolers.

The SW System satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Two SW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post-accident heat loads, assuming the worst single active failure occurs coincident with the loss of offsite power. Additionally, this system will also operate assuming the worst case passive failure post-RAS.

An SW subsystem is considered OPERABLE when:

a. The associated pump is OPERABLE; and

b. The associated piping, valves, heat exchangers, and instrumentation and controls required to perform the safety-related function are OPERABLE.

APPLICABILITY

In MODEs 1, 2, 3, and 4, the SW System is a normally operating system, which is required to support the OPERABILITY of the equipment serviced by the SW System and required to be OPERABLE in these MODEs.

In MODEs 5 and 6, the OPERABILITY requirements of the SW System are determined by the systems it supports.

ACTIONS

A.1

With one SW subsystem inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SW subsystem is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the SW subsystem

could result in loss of SW System function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions of LCO 3.8.1 should be entered if the inoperable SW subsystem results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6 should be entered if an inoperable SW subsystem results in an inoperable SDC. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

B.1 and B.2

If the SW subsystems cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the SW System flow path ensures that the proper flow paths exist for SW System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This surveillance test does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR is modified by a Note indicating that the isolation of the SW System components or systems may render those components inoperable but does not affect the OPERABILITY of the SW System.

BASES

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the SW System valves on an actual or simulated actuation signal (SIAS). The SW System is a normally operating system that cannot be fully actuated as part of the normal testing. This surveillance test is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint. Note: There are currently no SW valves with an Engineered Safety Feature Actuation System signal since automatic system reconfiguration during a LOCA is not required.

SR 3.7.7.3

The SR verifies proper automatic operation of the SW System pumps on an actual or simulated actuation signal (SIAS). The SW System is a normally operating system that cannot be fully actuated as part of the normal testing during normal operation. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 9.5.2.3, "Saltwater System"

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources-Operating

BASES

BACKGROUND

The AC sources to the Class 1E Electrical Power Distribution System consist of the offsite power sources starting at the 4.16 kV engineered safety feature (ESF) buses and the onsite diesel generators (DGs). As required by Reference 1, General Design Criteria (GDC) 17, the design of the AC electrical power system has sufficient independence and redundancy to ensure a source to the ESFs assuming a single failure.

The Class 1E AC Distribution System is divided into two redundant load groups so that the loss of one group does not prevent the minimum safety functions from being performed. Each load group has connections to two offsite sources and one Class 1E DG at its 4.16 kV 1E bus.

Offsite power is supplied to the 500 kV Switchyard from the transmission network by three 500 kV transmission lines. Two electrically and physically separated circuits supply electric power from the 500 kV Switchyard to two 13 kV buses and then to the two 4.16 kV ESF buses. A third 69 kV/13.8 kV offsite power source that may be manually connected to either 13 kV bus is available from the Southern Maryland Electric Cooperative (SMECO). When appropriate, the Engineered Safety Feature Actuation System (ESFAS) loss of coolant incident and shutdown sequencer for the 4.16 kV bus will sequence loads on the bus after the 69 kV/13.8 kV SMECO line has been manually placed in service. The SMECO offsite power source will not be used to carry loads for an operating unit. A detailed description of the offsite power network and the circuits to the Class 1E ESF buses, is found in Reference 2, Chapter 8.

The required offsite power circuits are the two 13 kV buses (Nos. 11 and 21) which can be powered by:

a. Two 500 kV lines, two 500 kV buses each of which have connections to a 500 kV line that does not pass through the other 500 kV bus and both P-13000 (500 kV/14 kV) transformers; or

b. One 500 kV line, one 500 kV bus, and one associated P-13000 (500 kV/13.8 kV) transformer, and the 69 kV/13.8 kV SMECO line. When the SMECO line is credited as one of the qualified offsite circuits, the disconnect from the SMECO line to Warehouse No. 1 must be open.

In addition, each offsite circuit includes the cabling to and from a 13.8/13.8 kV voltage regulator, 13.8/4.16 kV unit service transformer, and one of the two breakers to one 4.16 kV ESF bus. Transfer capability between the two required offsite circuits is by manual means only. The required circuit breaker to each 4.16 kV ESF bus must be from different 13.8/4.16 kV unit service transformers for the two required offsite circuits. Thus, each unit is able to align one 4.16 kV bus to one required offsite circuit, and the other 4.16 kV bus to the other required offsite circuit.

In some cases, inoperable components in the electrical circuit place both units in Conditions. Examples of these are 13.8 kV bus Nos. 11 or 21, two 500 kV transmission lines, one P-13000 service transformer, or one 500 kV bus. In other cases, inoperable components only place one unit in a Condition, such as an inoperable U-4000 and/or 13.8 kV regulator that feeds a required 4.16 kV bus.

The onsite standby power source to each 4.16 kV ESF bus is a dedicated DG. A DG starts automatically on an safety injection actuation signal or on a 4.16 kV degraded or undervoltage signal. If both 4.16 kV offsite source breakers are open, the DG, after reaching rated voltage and frequency, will automatically close onto the 4.16 kV bus.

In the event of a loss of offsite power to a 4.16 kV 1E bus, if required, the ESF electrical loads will be automatically sequenced onto the DG in sufficient time to provide for safe shutdown for an anticipated operational occurrence (AOO) and to ensure that the containment integrity and other vital functions are maintained in the event of a design bases accident.