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December 19, 2002

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, Revision 13

Enclosed for your use is one copy of the Calvert Cliffs Technical Specifications Bases, Revision 13. This revision was 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)."

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,

Peter E. Katz

PEK/DLM/bjd

Enclosures: As stated

cc: **(Without Enclosures)**
J. Petro, Esquire
J. E. Silberg, Esquire
Director, Project Directorate I-1, NRC
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A001

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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
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irradiated fuel assemblies in the spent fuel pool." Therefore, this LCO can be applicable in any or all MODEs. If the LCO and the Required Actions of LCO 3.7.13 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.13 of "Suspend movement of irradiated fuel assemblies in spent fuel pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

Limiting Condition for Operation 3.0.4 establishes limitations on changes in MODEs (including MODEs within the Applicability) or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit this Applicability to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODEs or other specified conditions in the Applicability

that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODEs or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. The exceptions allow entry into MODEs or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

Limiting Condition for Operation 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODEs 5 and 6 or in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4), because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Surveillance Requirements do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODEs or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those SRs that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

Limiting Condition for Operation 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 [e.g., to not comply with the

applicable Required Action(s)] to allow the performance of required testing to demonstrate:

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

b. The OPERABILITY of other equipment.

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

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

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

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

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

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

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if a loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross-train checks to identify a loss of safety function for those support systems supporting multiple and redundant safety systems are required. The cross-train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is

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with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3 Surveillance Requirement 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance test has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from when it is discovered that the surveillance test has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

This delay period provides an adequate time to complete missed surveillance tests. This delay period permits the completion of a surveillance test before complying with Required Actions or other remedial measures that might preclude completion of the surveillance test.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the surveillance test, the safety significance of the delay in completing the surveillance test, and the recognition that the most probable result of any particular surveillance test being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR Part 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the

missed Surveillance should be performed at the first reasonable opportunity.

Surveillance Requirement 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility that is not intended to be used as an operational convenience to extend surveillance test intervals.

While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance, as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed

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Surveillances will be placed in the licensee's Corrective Action Program.

If a surveillance test is not completed within the allowed delay period, the equipment is considered inoperable or the variable is considered outside the specified limits, and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a surveillance test fails within the delay period, the equipment is inoperable or the variable is outside the specified limits, and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the surveillance test.

Completion of the surveillance test within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

Surveillance Requirement 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified Condition in the Applicability.

This Specification ensures system and component OPERABILITY requirements and variable limits are met before entry into MODEs, or other specified conditions in the Applicability, for which these systems and components ensure safe operation of the unit.

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

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states surveillance tests do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the

SR(s) to be performed is removed. Therefore, failing to perform the surveillance test(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODEs or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODEs or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODEs or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the SR, or both. This allows performance of surveillance tests when the prerequisite condition(s) specified in a surveillance test procedure requires entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a surveillance test. A surveillance test that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the surveillance test may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

Surveillance Requirement 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, or MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODEs 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODEs 5 and 6, or in other specified conditions of the Applicability (unless in MODEs 1, 2, 3, or 4), because the ACTIONS of

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individual Specifications sufficiently define the remedial measures to be taken.

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. Reactor Coolant System flow is maintained above design flow by LCO 3.4.1.

4. Pressurizer Pressure-High Trip

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. Containment Pressure-High Trip

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.

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 of the bistable (sensor module) or outright failure or drift of the measurement channel sufficient to exceed the tolerance allowed by the plant-specific setting analysis. 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 check that the sensor modules are functioning properly. If the actual trip setting is not within the Allowable Value or not functioning, the channel is inoperable and the appropriate Conditions must be entered.

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 an alternate source.

16, 17, 18, 19. Core Exit Temperature Monitor

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.

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frequent, checks of channel during normal operational use of the displays associated with this LCO's required channels.

SR 3.3.10.2

A CHANNEL CALIBRATION is performed every 46 days on a staggered test basis for the Containment Hydrogen Analyzers. The CHANNEL CALIBRATION is performed using sample gases in accordance with manufacturer's recommendations.

SR 3.3.10.3

A CHANNEL CALIBRATION is performed every 24 months or approximately every refueling. CHANNEL CALIBRATION is a check of the indication channel including the sensor. The SR verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION of the CIV position indication channels will consist of verification that the position indication changes from not-closed to closed when the valve is exercised to the isolation position as required by Technical Specification 5.5.8, Inservice Testing Program. The position switch is the sensor for the CIV position indication channels. A Note allows exclusion of neutron detectors, Core Exit Thermocouples, and Reactor Vessel Level Monitor System from the CHANNEL CALIBRATION.

The Frequency is based upon operating experience and consistency with the typical industry refueling cycle and is justified by an 24 month calibration interval for the determination of the magnitude of equipment drift.

REFERENCES

1. Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated June 6, 1995, "License Amendment Request; Extension of Instrument Surveillance Intervals"

2. Letter from Mr. J. A. Tiernan (BGE) to NRC Document Control Desk, dated August 9, 1988, "Regulatory Guide 1.97 Review Update"

3. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident (Errata Published July 1981), December 1975

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4. NUREG-0737, Supplement 1, Requirements for Emergency Response Capabilities (Generic Letter 82-33), December 17, 1982

5. UFSAR, Chapter 7, "Instrumentation and Control"

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Reference 1) of normal operating conditions and anticipated operational occurrences, assume initial conditions within the normal steady-state envelope. The limits placed on departure from nucleate boiling (DNB) related parameters ensure that these parameters will not be less conservative than were assumed in the analyses, and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The Limiting Condition for Operation (LCO) limit for minimum RCS pressure as measured at the pressurizer is consistent with operation within the nominal operating envelope and is bounded by the initial pressure in the analyses.

The LCO limit for maximum RCS cold leg temperature is consistent with operation at the indicated power level and is bounded by the initial temperature in the analyses.

The LCO limit for minimum RCS flow rate is bounded by the initial flow rate in the analyses. The RCS flow rate is not expected to vary during plant operation with all pumps running.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the safety analyses (Reference 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed include loss of coolant flow events and dropped or stuck control element assembly events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, LCO 3.2.4, and LCO 3.2.5. The safety analyses are performed over the following range of initial

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values: RCS pressure 2154-2300 psia, core inlet temperature ≤ 548°F, and reactor vessel inlet coolant flow rate ≥ 370,000 gpm**.

The RCS DNB limits satisfy 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

This LCO specifies limits on the monitored process variables - RCS pressurizer pressure, RCS cold leg temperature, and RCS total flow rate - to ensure that the core operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

The LCO numerical values for pressure and temperature (P/T) are given for the measurement location and have been adjusted for instrument error. Reactor Coolant System flow rate is given as an analytical value.

APPLICABILITY

In MODE 1, the limits on RCS pressurizer pressure, RCS cold leg temperature, and RCS flow rate must be maintained during steady-state operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODEs, the power level is low enough so that DNBR is not a concern.

A Note has been added to indicate the limit on pressurizer pressure may be exceeded during short-term operational transients such as a THERMAL POWER ramp increase of > 5% RATED THERMAL POWER (RTP) per minute or a THERMAL POWER step increase of > 10% RTP. These conditions represent short-term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

** The Reactor Coolant System Total Flow Rate limit shall be ≥ 340,000 gpm through Unit 2, Cycle 14.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS LOOPS - MODEs 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process and transfer of this heat, via the SGs, to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron poison, boric acid;

d. Providing a second barrier against fission product release to the environment; and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The RCS configuration for heat transport uses two RCS loops. Each RCS loop contains a SG and two reactor coolant pumps (RCPs). An RCP is located in each of the two SG cold legs. The pump flow rate has been sized to provide core heat removal with appropriate margin to DNB during power operation and for anticipated transients originating from power operation. This Specification requires two RCS loops with both RCPs in operation in each loop. The intent of the Specification is to require core heat removal with forced flow during power operation. Specifying two RCS loops provides the minimum necessary paths (two SGs) for heat removal.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the DBA initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

Both transient and steady-state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming four RCPs are in operation. The majority of the plant

BASES

safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are of most importance to RCP operation are loss of coolant flow and seized rotor (Reference 1).

RCS Loops - MODEs 1 and 2 satisfy 10 CFR 50.36(c)(2)(ii), Criteria 2 and 3.

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by having both RCS loops with both RCPs in each loop in operation for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power.

Each OPERABLE loop consists of two RCPs providing forced flow for heat transport to an SG that is OPERABLE in accordance with the Steam Generator Tube Surveillance Program. Steam generator, and hence RCS loop, OPERABILITY with regard to SG water level is ensured by the RPS in MODEs 1 and 2. A reactor trip places the plant in MODE 3 if any SG level is ≥ 50 inches below normal water level* as sensed by the RPS. The minimum water level to declare the SG OPERABLE is < 50 inches below normal water level*.

APPLICABILITY

In MODEs 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE, and in operation in these MODEs to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODEs as indicated by the LCOs for MODEs 3, 4, 5, and 6.

Operation in other MODEs is covered by: LCO 3.4.5, LCO 3.4.6, LCO 3.4.7, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

* For Unit 2, the value shall remain 10 inches below the top of the feed ring through Cycle 14.

BASES

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs, and minimizes the possibility of violating DNB limits. It should be noted that the reactor will trip and place the plant in MODE 3 as soon as the RPS senses less than 370,000 gpm** RCS flow.

The Completion Time of six hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours of the required number of loops in operation. Verification includes flow rate, temperature, or pump status monitoring, which help to ensure that forced flow is providing heat removal while maintaining the margin to DNB. The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, Control Room indication and alarms will normally indicate loop status.

REFERENCES

1. UFSAR, Chapter 14, "Safety Analysis"

** The Reactor Coolant System Flow Rate limit shall be ≥ 340,000 gpm through Unit 2, Cycle 14.

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 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. 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

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setting range for each MSSV for OPERABILITY; however, the valves are reset to ± 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

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 (≥ 2800 ft for the steam-driven pump and ≥ 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 up to 24 hours 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 up to 24 hours 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 should be deferred up to 24 hours 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 up to 24 hours after reaching 800 psig in the steam generators. The Note ensures that proper test

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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 3.7.9 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.

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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. Therefore, 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 train of CRETS to be OPERABLE when 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.

B 3.7 PLANT SYSTEMS

B 3.7.15 Main Feedwater Isolation Valves (MFIVs)

BASES

BACKGROUND

The MFIVs isolate MFW flow to the secondary side of the steam generators following a HELB. The consequences of HELBs occurring in the main steam lines or in the MFW lines downstream of the MFIVs will be mitigated by their closure. Closure of the MFIVs effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for SLBs /or feedwater line breaks (FWLBs) inside the Containment Structure upstream of the reverse flow check valve, and reducing the cooldown effects for SLBs.

The MFIVs isolate the non-safety-related portions from the safety-related portion of the system. In the event of a secondary side pipe rupture inside the Containment Structure upstream of the reverse flow check valve, the valves limit the quantity of high energy fluid that enters the Containment Structure through the break.

One MFIV is located on each MFW line, outside, but close to, the Containment Structure. The MFIVs are located so that AFW may be supplied to a steam generator following MFIV closure. The piping volume from the valve to the steam generator must be accounted for in calculating mass and energy releases.

The MFIVs close on receipt of a steam generator isolation signal generated by low steam generator pressure. The steam generator isolation signal also actuates the MSIVs to close. The MFIVs may also be actuated manually. In addition, the MFIVs reverse flow check valve inside the Containment Structure is available to isolate the feedwater line penetrating the Containment Structure, and to ensure that the consequences of events do not exceed the capacity of the Containment Cooling System.

A description of the MFIVs operation on receipt of an steam generator isolation signal is found in Reference 1.

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs is established by the analysis for the large SLB. It is also influenced by the accident analysis for the large FWLB.

Failure of an MFIV to close following an SLB or FWLB can result in additional mass and energy to the steam generator's contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

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

LCO

This LCO ensures that the MFIVs will isolate MFW flow to the steam generators. Following an FWLB or SLB, these valves will also isolate the non-safety-related portions from the safety-related portions of the system. This LCO requires that one MFIV in each feedwater line be OPERABLE. The MFIVs are considered OPERABLE when the isolation times are within limits, and are closed on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to the Containment Structure following an SLB or FWLB inside the Containment Structure. Failure to meet the LCO can also add additional mass and energy to the steam generators contributing to cooldown.

APPLICABILITY

The MFIVs must be OPERABLE whenever there is significant mass and energy in the RCS and steam generators.

In MODEs 1, 2, and 3, the MFIVs are required to be OPERABLE in order to limit the amount of available fluid that could be added to the Containment Structure in the case of a secondary system pipe break inside the Containment Structure.

In MODEs 4, 5, and 6, steam generator energy is low.

perform the surveillance test, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. This Frequency is consistent with Reference 2, Chapter 8.

SR 3.8.1.14

This SR ensures that the manual synchronization and load transfer from the DG to the offsite source can be made and that the DG can be returned to ready-to-load status when offsite power is restored. The DG is considered to be in ready-to-load status when the DG is at rated speed and voltage, the output breaker is open and can receive an auto-close signal on bus undervoltage, and the load sequence timers are reset.

The Frequency of 24 months takes into consideration unit conditions required to perform the surveillance test.

SR 3.8.1.15

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This SR demonstrates the DG operation during a loss of offsite power actuation test signal in conjunction with an ESF (i.e., safety injection) actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

It is not necessary to energize loads which are dependent on temperature to load (i.e., heat tracing, switchgear HVAC compressor, computer room HVAC compressor). Also, it is acceptable to transfer the instrument AC bus to the non tested train to maintain safe operation of the plant during

testing. Loads (both permanent and auto connect) < 15 kW do not require loading onto the diesel since these are insignificant loads for the DG.

Permanently- and auto-connected loads to the emergency diesel generators are defined as follows:

Permanently-Connected Load – Equipment that is not shed by an undervoltage or safety injection actuation signal and is normally operating, i.e., loads that are manually started, selected, or process signal controlled are not considered permanently-connected loads.

Auto-Connected Loads – Emergency equipment required for mitigating the events described in UFSAR Chapter 14 that are energized by loss-of-coolant incident sequencer actions after step zero and within the first minute of emergency diesel generator operation after the initiation of an undervoltage signal.

The Frequency of 24 months takes into consideration unit conditions required to perform the surveillance test and is intended to be consistent with an expected fuel cycle length of 24 months.

This SR is modified by a Note. The reason for the Note is to minimize mechanical wear and stress on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs.

SR 3.8.1.16

This SR lists the SRs that are applicable to the LCO 3.8.1.c (SRs 3.8.1.1, 3.8.1.2, 3.8.1.3, 3.8.1.5, 3.8.1.6, and 3.8.1.7). Performance of any SR for the LCO 3.8.1.c will satisfy both Unit 1 and Unit 2 requirements for those SRs. Surveillance Requirements 3.8.1.4, 3.8.1.8, 3.8.1.9, 3.8.1.10, 3.8.1.11, 3.8.1.12, 3.8.1.13, 3.8.1.14, and 3.8.1.15, are not required to be performed for the LCO 3.8.1.c. Surveillance Requirement 3.8.1.10 is not

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required because this SR verifies manual transfer of AC power sources from the normal offsite circuit to the alternate offsite circuit, but only one qualified offsite circuit is necessary for the LCO 3.8.1.c. Surveillance Requirements 3.8.1.4, 3.8.1.11, and 3.1.8.12 are not required because they are tests that deal with loads. Surveillance Requirement 3.8.1.8 verifies the interval between sequenced loads. Surveillance Requirement 3.8.1.14 verifies the proper sequencing with offsite power. Surveillance Requirement 3.8.1.9 verifies that the DG starts within 10 seconds. These SRs are not required because they do not support the function of the LCO 3.8.1.c to provide power to the CREVS, CRETS, and H_2 Analyzer. Surveillance Requirements 3.8.1.13 and 3.8.1.15 are not required to be performed because these SRs verify the emergency loads are actuated on an ESFAS signal for the Unit in which the test is being performed. The LCO 3.8.1.c DG will not start on an ESFAS signal for this Unit.

REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"
2. UFSAR
3. Regulatory Guide 1.9, Revision 3, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants," July 1993
4. Safety Guide 9, Revision 0, March 1971
5. NRC Safety Evaluation for Amendment Nos. 19 and 5 for Calvert Cliffs Nuclear Power Plant Unit Nos. 1 and 2, dated January 14, 1977
6. Regulatory Guide 1.93, Revision 0, "Availability of Electric Power Sources," December 1974
7. Generic Letter 84-15, Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability, July 2, 1984
8. Regulatory Guide 1.137, Revision 1, "Fuel-Oil Systems for Standby Diesel Generators," October 1979

9. Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. C. H. Cruse (BGE), dated April 2, 1996, Issuance of Amendments for Calvert Cliffs Nuclear Power Plant, Unit 1 (TAC No. M94030) and Unit 2 (TAC No. M94031)

10. IEEE Standard 308-1991, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations"

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 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. The boron concentration limit specified in the COLR includes an uncertainty allowance.

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 filtered or unfiltered 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 requires that the door is capable of being closed, that the door is unblocked, and no cables or hoses are run through the doorway. OPERABILITY of the containment outage door requires that the door is capable of being closed, that the door is unblocked (i.e., capable of being closed within 30 minutes), and no cables and hoses are run through the doorway. Containment outage door grating or truck ramps may be installed if the grating or truck ramps 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.

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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 through a filtered or unfiltered pathway 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.

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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 through a filtered or unfiltered pathway (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

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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 through a filtered or unfiltered pathway 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.

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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

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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 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"

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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.

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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 providing direct access from containment atmosphere to the outside atmosphere through a filtered or unfiltered pathway 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"

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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 through a filtered or unfiltered pathway 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 through a filtered or unfiltered pathway 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.

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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