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October 21, 2003

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 14

Enclosed for your use is one copy of the Calvert Cliffs Technical Specifications Bases, Revision 14. 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,

for
George Vanderheyden
Vice President - Calvert Cliffs Nuclear Power Plant

GV/DLM/dlm

Enclosures: As stated

cc: **(Without Enclosures)**
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A001

PAGE REPLACEMENT INSTRUCTIONS
Calvert Cliffs Nuclear Power Plant

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0		May 4, 1998
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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
13	September 5, 2002	December 19, 2002
14	May 14, 2003	

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.

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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^T_{xy}), Total Integrated Radial Peaking Factor (F^T_r), 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. At high power, the detector alarms maintain the peak LHR below the LHR LCO limit based on the LOCA analysis only. At low power, the non-LOCA LHR LCO limits are more restrictive. Operation within the axial shape index limits of the excore DNB LCO assure that these non-LOCA LHR LCO limits will not be reached. 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

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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

BASES

(Reference 2). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zirconium-water reaction.

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_{xy}^{T}, F_{r}^{T}, 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.

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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 $\leq$ 20% RTP. Reducing THERMAL POWER to $\leq$ 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.

BASES

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.

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"

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 Zirconium-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_{xy}^{T} 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_{xy}^{T} 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_{xy}^{T} exceeds its basic limitation (F_{xy}^{T} > all rods out, full power limit), six hours is provided to restore F_{xy}^{T} to within limits. The combination of THERMAL POWER and F_{xy}^{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_{xy}^{T} to within its limit is reasonable

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 Zirconium-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

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 process 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 analysis (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 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 Zirconium-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 analyses.

Fuel cladding damage does not normally occur while the reactor is operating at conditions outside these LCOs during otherwise normal operation. Fuel cladding damage could result, however, if an accident or AOO occurs from initial conditions outside the limits of these LCOs. Changes in the power distribution cause increased power peaking and correspondingly increased local LHRs.

BASES

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

LCO

The power distribution LCO limits are based on correlations between power peaking and the 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 limits on LHR ensure that in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

APPLICABILITY

In MODE 1 with THERMAL POWER > 50% RTP, T_q 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 THERMAL POWER is not sufficient to require a limit on T_q.

ACTIONS

A.1 and A.2

If the measured T_q is > 0.03 and < 0.10, the calculation of T_q may be nonconservative. T_q must be restored within 4 hours, or F^T_{xy} and F^T_r must be determined to be within the limits of LCOs 3.2.2 and 3.2.3 within 4 hours, and determined to be within these limits every 8 hours thereafter, as long as T_q is out-of-limits. Four hours is sufficient time to allow the operator to reposition CEAs, and significant radial xenon redistribution cannot occur within this time. The 8 hour Completion Time ensures changes in F^T_{xy} and F^T_r can be identified before the limits of LCOs 3.2.2 and 3.2.3, respectively, are exceeded.

B.1

With T_q > 0.10, it must be restored to ≤ 0.10 with 2 hours. F^T_{xy} and F^T_r must be verified to be within their specified limits to ensure that acceptable flux peaking factors are maintained. Operation may proceed for a total of 2 hours, after the Condition is entered, while attempts are made to restore T_q to within its limit.

If the tilt is generated due to a CEA misalignment, operating at ≤ 50% RTP allows for the recovery of the CEA. Except as a result of CEA misalignment, T_q > 0.10 is not

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 Zirconium-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.

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. Unit 1 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. Manual Actuation

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

these Category I variables are important in reducing public risk.

LCO

Limiting Condition for Operation 3.3.10 requires two OPERABLE indication channels for all but one Function to ensure no single failure prevents the operators from being presented with the information necessary to determine the status of the plant and to bring the plant to, and maintain it in, a safe condition following that accident.

Furthermore, provision of two indication channels allows a CHANNEL CHECK during the post-accident phase to confirm the validity of displayed information.

An indication channel consists of field transmitters or process sensors and associated instrumentation, providing a measurable electronic signal based upon the physical characteristics of the parameter being measured, plus a display of the measured parameter.

The exceptions to the two-channel requirement are CIV position and the subcooled margin monitoring (SMM) instrumentation. In the case of valve position, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active CIV. This is sufficient to redundantly verify the isolation status of each isolable penetration, either via indicated status of the active valve and prior knowledge of the passive valve, or via system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE. Alternate means are available for obtaining information provided by the SMM instrumentation.

Listed below are discussions of the specified instrument Functions listed in Table 3.3.10-1.

1. <u>Wide Range Logarithmic Neutron Flux Monitors</u>

   Wide range logarithmic neutron flux is a Category I variable indication is provided to verify reactor shutdown.

The wide range logarithmic neutron flux PAM channels consist of two wide range neutron monitoring channels.

2, 3. RCS Outlet and Inlet Temperature

Reactor Coolant System outlet and inlet temperatures are Category I variables provided for verification of core cooling and long-term surveillance.

Reactor outlet temperature inputs to the PAM are provided by four resistance elements and associated transmitters in each loop. The channels provide indication over a range of 50°F to 700°F.

4. SMM

Unit 1 only

The RCS SMM is provided to monitor for inadequate core cooling by calculating the margin to saturation based on the RCS pressure/temperature relationships and displaying the calculated margin (1°F to 100°F) on a control room indicator. The SMM also generates a low subcooled margin alarm should the temperature margin drop below predetermined limits. The SMM is a microprocessor based instrument provided with inputs from the RCS hot and cold legs temperature instrumentation and wide range RCS pressure channels.

The RCS SMM is one of three components of inadequate core cooling instrumentation with the RCS SMM inoperable, the core exit thermocouple (CET) and reactor vessel water level monitoring systems provide diverse indication of core cooling.

Unit 2 only

The RCS SMM is part of the PAM System and is provided to monitor inadequate core cooling by calculating the margin to saturation based on the RCS pressure/ temperature relationships and displaying the calculated margin in degrees F on a control room display. Also, a control room low subcooled margin alarm is provided. The RCS SMM portion of the PAM System is

microprocessor-based and is provided with inputs from the RCS hot legs, cold legs, and wide range RCS pressure channels. The CET SMM and upper head SMM functions are not required for channel operability.

The RCS SMM is one of three components of inadequate core cooling instrumentation. With the SMM portion of the PAM System inoperable, the CETs and the reactor vessel water level heated junction thermocouple (HJTC) sensors provide diverse indication of core cooling. Alternate indications and methods for calculating subcooled margin exist in the vent of a PAM System failure.

5. Reactor Vessel Water Level

Reactor vessel water level indication is provided for verification and long-term surveillance testing of core cooling.

This indication uses a HJTC technology. This technology measures reactor coolant inventory with discrete heated junction thermocouple sensors located in different levels within a separator tube. The sensors enable a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory. The collapsed level is obtained over the same temperature and pressure range as the saturation measurements, thereby encompassing all operating and accident conditions where it must function. Also, it functions during the recovery interval. Therefore, it is designed to survive the high steam temperature that may occur during the preceding core recovery interval.

The level range extends from the top of the vessel down to 10" above the top of the fuel alignment plate. The response time is short enough to track the level during small break LOCA events. The resolution is sufficient to show the initial level drop, the key locations near the hot leg elevation, and the lowest levels just above

the alignment plate. This provides the operator with adequate indication to track the progression of the accident and to detect the consequences of its mitigating actions or the functionality of automatic equipment.

A channel has eight sensors in a probe. A channel is OPERABLE if four sensors, one in the upper three and three in the lower five, are OPERABLE.

6. Containment Sump Water Level (wide range) Monitor

Containment sump water level monitors are provided for verification and long-term surveillance of RCS integrity.

Containment sump water level instrumentation consists of two level transmitters that provide input to control room indicators. The transmitters are located above the containment flood level and utilize sealed reference legs to sense water level.

7. Containment Pressure (wide range) Monitor

The containment pressure monitor is provided for verification of RCS and containment OPERABILITY.

Containment pressure instrumentation consists of three containment pressure transmitters with overlapping ranges that provide input to control room indicators. The transmitters are located outside the Containment and are not subject to a harsh environment.

8. CIV Position Indicator

Containment isolation valve position indicators are provided for verification of containment OPERABILITY and integrity.

In the case of CIV position, the important information is the isolation status of the containment penetration. The LCO requires one channel of valve position indication in the Control Room to be OPERABLE for each active CIV in a containment penetration flow path, i.e., two total channels of CIV position indication for

a penetration flow path with two active valves. For containment penetrations with only one active CIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration via indicated status of the active valve, as applicable, and prior knowledge of passive valve or system boundary status. If a penetration flow path is isolated, position indication for the CIV(s) in the associated penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration flow path is not required to be OPERABLE.

The CIV position PAM instrumentation consists of ZL-505, 506, 515, 516, 2080, 2180, 2181, 3832, 3833, 4260, 5291, 5292, 6900, and 6901 (Reference 5).

9. <u>Containment Area Radiation (high range) Detector</u>

Containment area radiation detectors are provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operations in determining the need to invoke site emergency plans.

Containment area radiation instrumentation consists of two radiation detectors with displays and alarm in the Control Room. The radiation detectors have a measurement range of 1 to 10^8 R/hr.

10. <u>Containment Hydrogen Monitors</u>

Containment hydrogen monitors are provided to detect high hydrogen concentration conditions that represent a potential for Containment breach. This variable is also important in verifying the adequacy of mitigating actions.

Containment hydrogen instrumentation monitors samples from six locations inside the Containment. Two groups of three sampling lines from each Containment provide samples to the two cabinets in the sampling room. The

cabinets contain the hydrogen analyzers and related equipment. The sampling system is fully operational from this remote station due to post-accident personnel safety considerations. Once the system is placed in the sampling mode, any malfunction or high hydrogen condition will generate a signal in the annunciator system in the main Control Room.

11. Pressurizer Pressure (wide range)

Pressurizer wide range pressure is a Category I variable provided for verification of core cooling and RCS integrity long-term surveillance.

Wide range pressurizer pressure is measured by two pressure transmitters with a span of 0 psia to 4000 psia. The pressure transmitters are located inside the Containment. Redundant monitoring capability is provided by two indication channels. Control Room indications are provided.

Pressurizer pressure is a Type I variable because the operator uses this indication to monitor the cooldown of the RCS following a LOCA and other DBAs. Operator actions to maintain a controlled cooldown, such as adjusting steam generator pressure or level, would use this indication. Furthermore, pressurizer pressure is one factor that may be used in decisions to terminate RCP operation.

12. Steam Generator Pressure Transmitter

Steam generator pressure transmitters are Category 1 instruments and are provided to monitor operation of decay heat removal via the steam generators.

There are four redundant pressure transmitters per steam generator, but only two per steam generator are required to satisfy the Technical Specification Requirements. The transmitter provides wide range indication over the range from 0 to 1200 psia. Each transmitter provides input to control room indication. Since the primary indication used by the operator during an accident is the control room indicator, the

PAM instrumentation Specification deals specifically with this portion of the instrument channel.

13. Pressurizer Level Transmitters

Pressurizer level transmitters are used to determine whether to terminate safety injection, if still in progress, or to reinitiate safety injection if it has been stopped. Knowledge of pressurizer water level is also used to verify the plant conditions necessary to establish natural circulation in the RCS and to verify that the plant is maintained in a safe shutdown condition.

Pressurizer Level instrumentation consists of two pressurizer level transmitters that provide input to control room indicators.

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

Core Exit Temperature indication is provided for verification and long-term surveillance of core cooling.

An evaluation was made of the minimum number of valid CETs necessary for inadequate core cooling detection. The evaluation determined the reduced complement of CETs 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 CETs per quadrant.

The design of the Incore Instrumentation System includes a Type K (chromel alumel) thermocouple within each of the 35 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 CETs monitor the temperature of the reactor coolant as it exits the fuel assemblies.

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

20. Pressurizer Pressure (low range)

Pressurizer low range pressure is a Category I variable provided for verification of core cooling and RCS integrity long-term surveillance.

Low-range pressurizer pressure is measured by two pressure transmitters with a span of 0 psia to 1600 psia. The pressure transmitters are located inside the Containment. Redundant monitoring capability is provided by two indication channels. Control Room indications are provided.

Pressurizer pressure is a Type I variable because the operator uses this indication to monitor the cooldown of the RCS following a LOCA and other DBAs. Operator actions to maintain a controlled cooldown, such as adjusting steam generator pressure or level, would use this indication. Furthermore, pressurizer pressure is one factor that may be used in decisions to terminate RCP operations.

Two indication channels are required to be OPERABLE for all but two Functions. Two OPERABLE channels ensure that no single failure, within either the PAM instrumentation or its auxiliary supporting features or power sources (concurrent with the failures that are a condition of or result from a specific accident), prevents the operators from being presented the information necessary for them to determine

the safety status of the plant, and to bring the plant to and maintain it in a safe condition following that accident.

In Table 3.3.10-1 the exceptions to the two channel requirement are CIV position and the SMM.

Two OPERABLE CETs are required for each channel in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Therefore, two randomly selected thermocouples may not be sufficient to meet the two thermocouples per channel requirement in any quadrant. The two thermocouples in each channel must meet the additional requirement that one be located near the center of the core and the other near the core perimeter, such that the pair of CETs indicate the radial temperature gradient across their core quadrant. The two channels in each core quadrant must be electronically independent. A CETs operability is based on a comparison of the CET temperature indication with the hot leg resistance temperature detector temperature indication. Different criteria have been specified for interior CETs and peripheral CETs to account for the core radial power distribution. Plant specific evaluations in response to Item II.F.2 of NUREG-0737 should have identified the thermocouple pairings that satisfy these requirements. Two sets of two thermocouples in each quadrant ensure a single failure will not disable the ability to determine the radial temperature gradient.

For loop- and steam generator-related variables, the required information is individual loop temperature and individual steam generator level. In these cases, two channels are required to be OPERABLE for each loop of steam generator to redundantly provide the necessary information.

In the case of CIV position, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active CIV. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of the passive valve or via

system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

The SMM, CETs, and the HJTC-based reactor vessel water level indication comprise the inadequate core cooling instrumentation. The function of the inadequate core cooling instrumentation is to enhance the ability of the plant operator to diagnose the approach to, and recovery from, inadequate core cooling.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODEs 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODEs 1, 2, and 3. In MODEs 4, 5, and 6, plant conditions are such that the likelihood of an event occurring requiring PAM instrumentation is low; therefore, PAM instrumentation is not required to be OPERABLE in these MODEs.

ACTIONS

Note 1 has been added in the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS, even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the passive function of the indication channels, the operator's ability to monitor an accident using alternate instruments and methods, and the low probability of an event requiring these indication channels.

Note 2 has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.10-1. 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.

<u>A.1</u>

When one or more Functions have one required indication channel that is inoperable, the required inoperable channel

must be restored to OPERABLE status within 30 days. The 30-day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Reference 3 indication channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

<u>B.1</u>

This Required Action specifies initiation of actions in accordance with Specification 5.6.7, which requires a written report to be submitted to the NRC. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative Required Actions. This Required Action is appropriate in lieu of a shutdown requirement, given the likelihood of plant conditions that would require information provided by this instrumentation. Also, alternative Required Actions such as grab sampling or diverse indications are identified before a loss of functional capability condition occurs.

<u>C.1</u>

When one or more Functions have two required indication channels inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

D.1

When two required hydrogen monitor channels are inoperable, Required Action D.1 requires one channel to be restored to OPERABLE status. This Required Action restores the monitoring capability of the hydrogen monitor. The 72-hour Completion Time is based on the relatively low probability of an event requiring hydrogen monitoring and the availability of alternative means to obtain the required information. Continuous operation with two required channels inoperable is not acceptable because alternate indications are not available.

E.1

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.10-1. The applicable Condition referenced in the Table is Function-dependent. Each time Required Action C.1 or D.1 is not met and the associated Completion Time has expired, Condition E is entered for that channel and provides for transfer to the appropriate subsequent Condition.

F.1 and F.2

If the Required Action and associated Completion Time of Condition C are not met, and Table 3.3.10-1 directs entry into Condition F, the plant must be brought to a MODE in which the requirements of this LCO do not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to 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.

G.1

Alternate means of monitoring containment area radiation have been developed and tested. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. The HJTC-based reactor vessel water level instrumentation is one of three components of the inadequate core cooling instrumentation. The SMM instrumentation and CETs could be

used to monitor inadequate core cooling. If these alternate means are used, the Required Action is not to shut down the plant, but rather to follow the directions of Specification 5.6.7. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

SURVEILLANCE REQUIREMENTS

A Note at the beginning of the SRs specifies that the following SRs apply to each PAM instrumentation Function in Table 3.3.10-1.

SR 3.3.10.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one indication channel to a similar parameter on other channels. It is based on the assumption that indication channels monitoring the same parameter should read approximately the same value. Significant deviations between the two indication channels could be an indication of excessive instrument drift in one of the channels or of something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a qualitative assessment of the indication channel that considers indication channel uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off-scale during times when surveillance testing is required, the CHANNEL CHECK will only verify that they are off-scale in the same direction. Off-scale low current loop channels are verified to be reading at the bottom of the range and not failed down-scale.

For the Hydrogen Monitors, a CHANNEL CHECK is performed by drawing a sample from the Waste Gas System through the monitor.

The Frequency of 31 days is based upon plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one indication channel of a given Function in any 31 day interval is a rare event. The CHANNEL CHECK supplements less formal, but more 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, CETs, and reactor vessel level (HJTC) 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.

BASES

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

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

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

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

Six months is reasonable for verification by sampling 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. This Frequency is adequate to identify changes that could occur from mechanisms, such as stratification or inleakage.

Verification consists of monitoring inleakage or sampling. The inleakage is monitored every 12 hours by monitoring tank level. Sampling of each tank is done every six months. All intentional sources of level increase are maintained administratively to ensure SIT boron concentrations are within technical specification limits. The boron concentration of each tank is verified prior to startup from outages. 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

BASES

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 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"

B 3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

B 3.5.4 Refueling Water Tank (RWT)

BASES

BACKGROUND

The RWT supports the ECCS and the Containment Spray System by providing a source of borated water for ESF pump operation.

The RWT supplies two ECCS trains by separate, redundant supply headers. Each header also supplies one train of the Containment Spray System. A motor-operated isolation valve is provided in each header, to allow the operator to isolate the usable volume of the RWT from the ECCS after the ESF pump suction has been transferred to the containment sump, following depletion of the RWT during a LOCA. A separate header is used to supply the Chemical and Volume Control System from the RWT. Use of a single RWT to supply both trains of the ECCS is acceptable, since the RWT is a passive component, and passive failures are not assumed to occur coincidentally with the Design Basis Event during the injection phase of an accident. Not all the water stored in the RWT is available for injection following a LOCA; the location of the ECCS suction piping in the RWT will result in some portion of the stored volume being unavailable.

The HPSI, LPSI, and containment spray pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at shutoff head conditions. These lines discharge back to the RWT, which vents to the atmosphere. When the suction for the HPSI and containment spray pumps is transferred to the containment sump, this flow path must be isolated to prevent a release of the containment sump contents to the RWT. If not isolated, this flow path could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ESF pumps.

This LCO ensures that:

a. The RWT contains sufficient borated water to support the ECCS during the injection phase;

b. Sufficient water volume exists in the containment sump to support continued operation of the ESF pumps at the

time of transfer to the recirculation mode of cooling; and

c. The reactor remains subcritical following a LOCA.

Insufficient water inventory in the RWT could result in insufficient cooling capacity of the ECCS when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside Containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWT provides a source of borated water to the HPSI, LPSI, containment spray, and charging pumps when level is low in the boric acid tanks. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory, and is a source of negative reactivity for reactor shutdown (Reference 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses Section of B 3.5.2 and B 3.6.6. These analyses are used to assess changes to the RWT in order to evaluate their effects in relation to the acceptance limits.

The volume limit of 400,000 gallons is based on two factors:

a. Sufficient deliverable volume must be available to provide at least 32 minutes (plus a 10% margin) of full flow from all ESF pumps prior to reaching a low level switchover to the containment sump for recirculation; and

b. The containment sump water volume must be sufficient to support continued ESF pump operation after the switchover to recirculation occurs. This sump volume water inventory is supplied by the RWT borated water inventory.

When ESF pump suction is transferred to the sump, there must be sufficient water in the sump to ensure adequate net positive suction head for the HPSI and containment spray pumps. The RWT capacity must be sufficient to supply this

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). Main steam isolation valve closure terminates flow from the unaffected (intact) steam generator.

One MSIV is located in each main steam line outside, but close to, the Containment Structure. The MSIVs are downstream from the MSSVs, atmospheric dump valves (ADVs), and AFW pump turbine steam supplies to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a steam generator isolation signal generated by low steam generator pressure or on a containment spray actuation signal (CSAS) generated by high containment pressure. The MSIVs fail closed on loss of control or actuation power. The steam generator isolation signal also actuates the main feedwater isolation valves (MFIVs) to close. The MSIVs may also be actuated manually.

A description of the MSIVs is found in Reference 1, Section 10.1.

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside the Containment Structure, as discussed in Reference 1, Section 14.20. It is also influenced by the accident analysis of the SLB events presented in Reference 1, Section 14.14. The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

The limiting case for main SLB Containment Structure response is 75% power, no loss of offsite power, and failure of a steam generator feed pump to trip. This case results in continued feeding of the affected steam generator and maximizes the energy release into the Containment Structure.

BASES

This case does not assume failure of an MSIV; however, an important assumption is both MSIVs are OPERABLE. This prevents blowdown of both steam generators assuming failure of an MSIV to close.

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside the Containment Structure upstream of the MSIV is the limiting SLB for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside the Containment Structure at hot full power is the limiting case for a post-trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip.

The MSIVs only serve a safety function and remain open during power operation. These valves operate under the following situations:

a. An HELB inside the Containment Structure. In order to maximize the mass and energy release into the Containment Structure, the analysis assumes steam is discharged into the Containment Structure from both steam generators until closure of the MSIV occurs. After MSIV closure, steam is discharged into the Containment Structure only from the affected steam generator.

b. A break outside of the Containment Structure and upstream from the MSIVs. This scenario is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs. This type of break will be isolated by the closure of the MSIVs. Events such as increased steam flow through the turbine or the steam bypass valves (e.g., excess load event) will also terminate on closure of the MSIVs.

d. A steam generator tube rupture. For this scenario, closure of the MSIV isolates the affected steam

generator from the intact steam generator and minimizes radiological releases. The operator is then required to maintain the pressure of the steam generator with the ruptured tube below the MSSV setpoints, a necessary step toward isolating the flow through the rupture.

e. The MSIVs are also utilized during other events such as a feedwater line break. These events are less limiting so far as MSIV OPERABILITY is concerned.

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

LCO

This LCO requires that the MSIV in each of the two steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents as described in Reference 1, Chapter 14.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1 and in MODEs 2 and 3, except when all MSIVs are closed. In these MODEs there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing their safety function.

In MODE 4, the steam generator energy is low; therefore, the MSIVs are not required to be OPERABLE.

In MODEs 5 and 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODEs.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, time is allowed to restore the component to OPERABLE status. Some repairs can be made to the MSIV with the unit hot. The eight hour Completion Time is reasonable, considering the probability of an accident occurring during the time period that would require closure of the MSIVs.

B.1

If the MSIV cannot be restored to OPERABLE status within eight hours, 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 MODE 2 within six hours and Condition C would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2, and close the MSIVs in an orderly manner and without challenging unit systems.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODEs 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The eight hour Completion Time is consistent with that allowed in Condition A.

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The seven day Completion Time is reasonable, based on engineering judgment, MSIV status indications available in the Control Room, and other administrative controls, to ensure these valves are in the closed position.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status, or closed, 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 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from

BASES

MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is < 5.2 seconds. The MSIV closure time is assumed in the accident and containment analyses.

The Frequency for this SR is in accordance with the Inservice Testing Program. The MSIVs are tested during each refueling outage in accordance with Reference 2, and sometimes during other cold shutdown periods. The Frequency demonstrates the valve closure time at least once per refueling cycle. Operating experience has shown that these components usually pass the SR when performed. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR
2. ASME, Boiler and Pressure Vessel Code, Section XI, 1989, "Rules for In-Service Inspection of Nuclear Power Plant Components," and ASME Operation and Maintenance Code Part 10, 1987, with 1988 Addenda

BASES

ACTIONS

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

A.1

With one MFIV inoperable, action must be taken to restore the valve to OPERABLE status within 72 hours.

The 72 hour Completion Time takes into account the isolation capability afforded by the MFW regulating valves, and tripping of the MFW pumps, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

B.1 and B.2

If the MFIVs cannot be restored to OPERABLE status in 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 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 3.7.15.1

This SR ensures the closure time for each MFIV is $\leq$ 65 seconds by manual isolation. The MFIV closure time is assumed in the accident and containment analyses.

The Frequency is in accordance with the Inservice Testing Program. The MFIVs are tested during each refueling outage in accordance with Reference 2, and sometimes during other cold shutdown periods. The Frequency demonstrates the valve closure time at least once per refueling cycle. Operating experience has shown that these components usually pass the surveillance test when performed.

BASES

REFERENCES	1.	UFSAR, Section 14.4.2, "Sequence of Events"
	2.	ASME, Boiler and Pressure Vessel Code, Section XI, 1989, "Rules for In-Service Inspection of Nuclear Power Plant Components," and ASME Operation and Maintenance Code Part 10, 1987, with 1988 Addenda

B 3.9 REFUELING OPERATIONS

B 3.9.5 Shutdown Cooling (SDC) and Coolant Circulation-Low Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and other residual heat from the RCS, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Reference 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s), where the heat is transferred to the Component Cooling Water System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the Control Room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the SDC System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to the resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and due to the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two loops of the SDC System are required to be OPERABLE, and one loop is required to 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 prevent this challenge.

Shutdown cooling and Coolant Circulation-Low Water Level satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

BASES

LCO

In MODE 6, with the water level < 23 ft above the top of the irradiated fuel assemblies seated in the reactor vessel, both SDC loops must be OPERABLE. Additionally, one loop of the SDC System must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE SDC loop consists of an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs. Both SDC pumps may be aligned to the Refueling Water Tank to support filling the refueling pool or for performance of required testing.

The LCO is modified by a note that allows one required SDC loop to be replaced by one spent fuel pool cooling loop when it is lined up to provide cooling flow to the irradiated fuel assemblies in the reactor core, and the heat generation rate of the core is below the heat removal capacity of the spent fuel cooling loop.

This LCO is modified by a Note that allows one SDC loop to be inoperable for a period of two hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing plant configuration. This consideration should include that the core time to boil is short, there is no draining operation to further reduce RCS water level and that the capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible.

This LCO is modified by a Note that permits the SDC pumps to be deenergized for ≤ 15 minutes when switching from one train to another. The circumstances for stopping both SDC pumps are to be limited to situations when the pump outage time is short and the core outlet temperature is maintained

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> 10°F below saturation temperature. The Note prohibits boron dilution or draining operations when SDC forced flow is stopped.

APPLICABILITY

Two SDC loops are required to be OPERABLE, and 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. Requirements for the SDC System in other MODEs are covered by LCOs in Section 3.4. MODE 6 requirements, with a water level ≥ 23 ft above the irradiated fuel assemblies seated in the reactor vessel, are covered in LCO 3.9.4.

ACTIONS

A.1 and A.2

If one SDC loop is inoperable, action shall be immediately initiated and continued until the SDC loop is restored to OPERABLE status and is in operation, or until the water level is ≥ 23 ft above the irradiated fuel assemblies seated in the reactor vessel. When the water level is established at ≥ 23 ft above the irradiated fuel assemblies 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

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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.

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 is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

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