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Richard J. St. Onge
Director
Nuclear Regulatory Affairs

February 22, 2011

10 CFR 50.4

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: **Docket No. 50-362**
Core Operating Limits Report
San Onofre Nuclear Generating Station, Unit 3

Reference: NRC letter from James R. Hall (NRC) to Ross T. Ridenoure, dated December 15, 2009, San Onofre Nuclear Generating Station, Units 2 and 3 - Issuance of Amendments Revising Technical Specification 5.7.1.5, "Core Operating Limits Report (COLR)" (TAC NOS. ME0604 and ME0605)

Dear Sir or Madam:

The Core Operating Limits Report (COLR) for Cycle 16 for the San Onofre Nuclear Generating Station (SONGS) Unit 3 is provided as an enclosure to this letter. This submittal is made in accordance with Section 5.7.1.5.d, "Core Operating Limits Report (COLR)," of the SONGS Unit 3 Technical Specifications.

The COLR is contained in the Unit-specific Licensee Controlled Specifications (LCS).

Consistent with the Reference, a Licensee Controlled Specification change was made to LCS Section 5.0.105, to add "Core Operating Limits Report (COLR) Analytical Methods" to include analytical method 9, "SCE-0901-A, PWR Reactor Physics Methodology Using Studsvik Design Codes."

This submittal contains no new commitments.

If you have any questions regarding this information, please contact Ms. Linda T. Conklin at (949) 368-9443.

Sincerely,

A001
NRR

P.O. Box 128
San Clemente, CA 92674

Enclosure

cc: E. E. Collins, Jr., Regional Administrator, NRC Region IV
R. Hall, NRC Project Manager, San Onofre Units 2 and 3
G. G. Warnick, NRC Senior Resident Inspector, San Onofre Units 2 and 3

Enclosure

Core Operating Limits Report (COLR)
Cycle 16
San Onofre Nuclear Generating Station
(SONGS) Unit 3

3.1 REACTIVITY CONTROL SYSTEMS

LCS 3.1.100 Moderator Temperature Coefficient (MTC)

The MTC shall be > [more positive than] -3.7 E-4 $\Delta k/k/°F$ at RTP.

AND

The steady state MTC shall be no more positive than the upper MTC limit shown in Figure 3.1.100-1.

VALIDITY STATEMENT: Effective upon start of Cycle 9.

APPLICABILITY: MODES 1 and 2 with $K_{eff} \geq 1.0$ except during PHYSICS TESTS under the Special Test Exemptions of the Technical Specifications.

ACTIONS

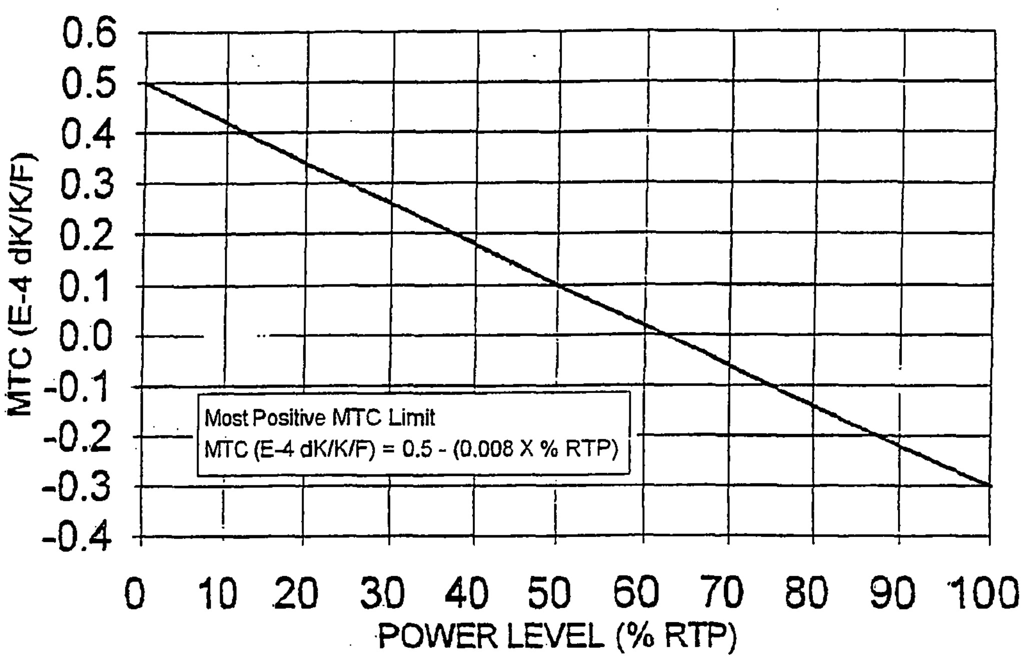
CONDITION	REQUIRED ACTION	COMPLETION TIME
	Refer to LCO 3.1.4	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Refer to LCO 3.1.4	

NOTE: Predicted MTC values shall be adjusted based on Mode 2 measurements to permit direct comparison with Figure 3.1.100-1.

Figure 3.1.100-1
MOST POSITIVE MTC VS. POWER



LCS 3.1.100 Moderator Temperature Coefficient (MTC)

BASES

The limitations on MTC are provided to ensure that the assumptions used in the the accident and transient analysis remain valid throughout each fuel cycle. The limiting events with respect to the MTC limits are; a CEA ejection at the beginning of core life and a main steam line break at the end of core life. The Surveillance Requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3.1 REACTIVITY CONTROL SYSTEMS

LCS 3.1.102 Regulating CEA Insertion Limits

The regulating CEA groups shall be limited to the withdrawal sequence, and insertion limits specified in Figure 3.1.102-1.

VALIDITY STATEMENT: Revisions 1 and 2 effective 02/12/99, to be implemented within 30 days.

APPLICABILITY: MODE 1 and 2 except during PHYSICS TESTS under the Special Test Exemptions of the Technical Specifications.

ACTIONS

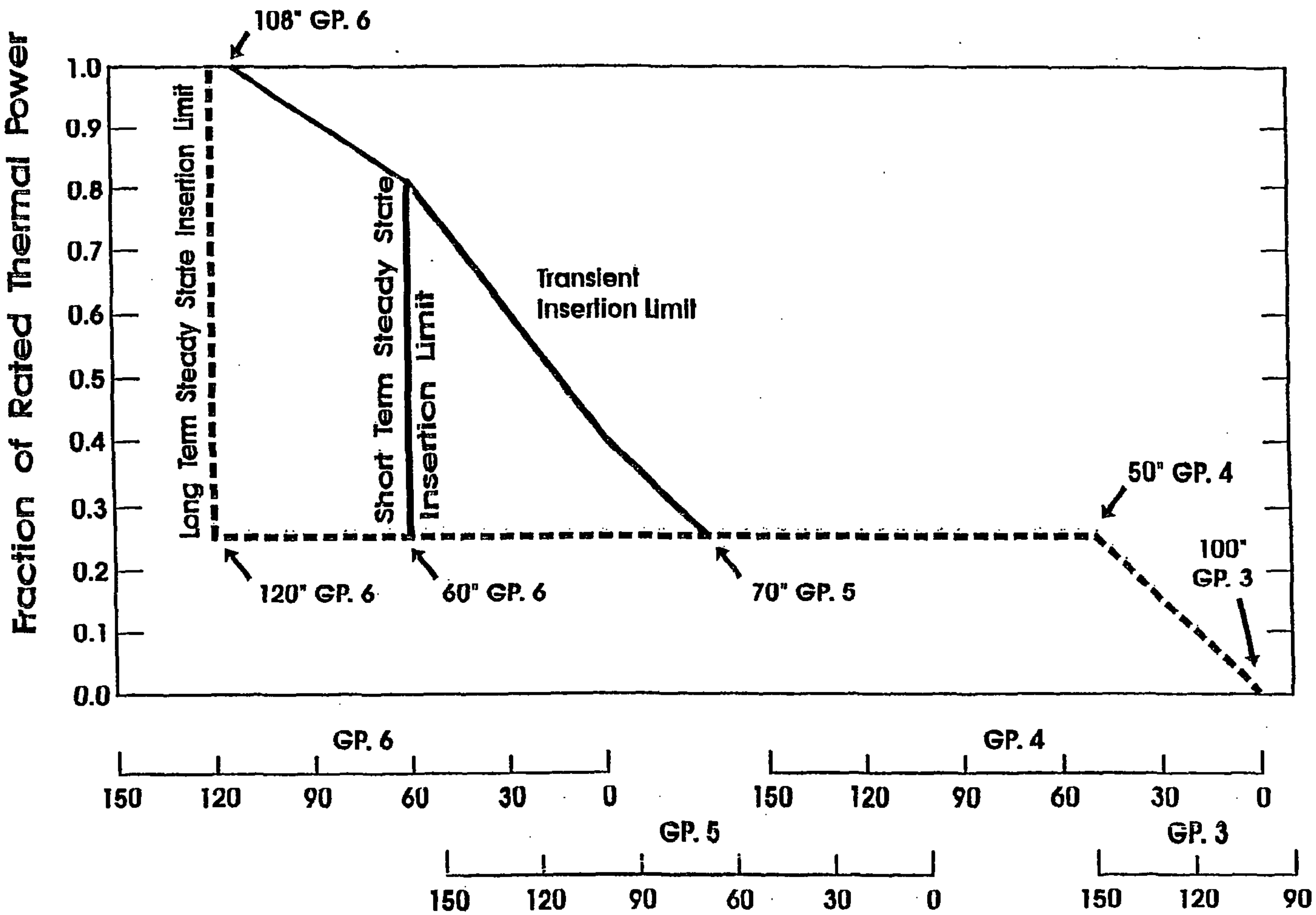
CONDITION	REQUIRED ACTION	COMPLETION TIME
	Refer to LCO 3.1.7	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Refer to LCO 3.1.7	

FIGURE 3.1.102-1

REGULATING CEA WITHDRAWAL VS THERMAL POWER



Critical CEA Position - Inches Withdrawn

LCS 3.1.102 Regulating CEA Insertion Limits

Bases

The Core Operating Limits Report (COLR) Licensee Controlled Specification (LCS) for Regulating Control Element Assembly (CEA) Insertion Limits provides CEA withdrawal sequence and insertion limits while operating in Modes 1 and 2. The long term and short term steady state insertion limits and transient insertion limits for each regulating CEA group are specified graphically as a function of the fraction of rated Thermal Power. These limits ensure that an acceptable power distribution and the minimum shutdown margin is maintained, and the potential effects of CEA misalignment are limited to an acceptable level. Limited deviations from the nominal requirements are permitted with Technical Specification (TS) ACTION statements providing additional compensatory restrictions and time limits. TS Surveillance Requirements provide assurance that necessary system components are OPERABLE and CEA group positions that may approach or exceed acceptable limits are detected, with adequate time for an Operator to take any required Action.

In Mode 2 with $K_{eff} < 1.0$, LCS Figure 3.1.102-1 still applies; for this condition the CEAs must be withdrawn sufficiently such that if the CEAs were to be (further) withdrawn to criticality ($K_{eff} = 1.0$) with no boration, then that critical CEA position would be further withdrawn than the position required by LCS Figure 3.1.102-1. This comparison is appropriate since it is the critical CEA position compared to the insertion limit of LCS Figure 3.1.102-1 that determines whether requirements are satisfied regarding shutdown margin and potential effects of CEA misalignment. Before criticality this condition is verified by selection of a critical CEA position and critical boron concentration that is calculated to show a critical rod position above the regulating CEA insertion limit at zero power. After criticality compliance is shown by critical position being above the regulating CEA insertion limit at zero power.

3.1 REACTIVITY CONTROL SYSTEMS

LCS 3.1.103 Part-Length CEA Insertion Limits

The Part-Length CEA groups shall be limited to the insertion limits specified in Figure 3.1.103-1.

VALIDITY STATEMENT: Effective upon TSIP Implementation.

APPLICABILITY: MODE 1 > 20% RTP except during PHYSICS TESTS under the Special Test Exemptions of the Technical Specifications.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
	Refer to LCO 3.1.8	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Refer to LCO 3.1.8	

PART LENGTH CEA INSERTION LIMIT VS THERMAL POWER

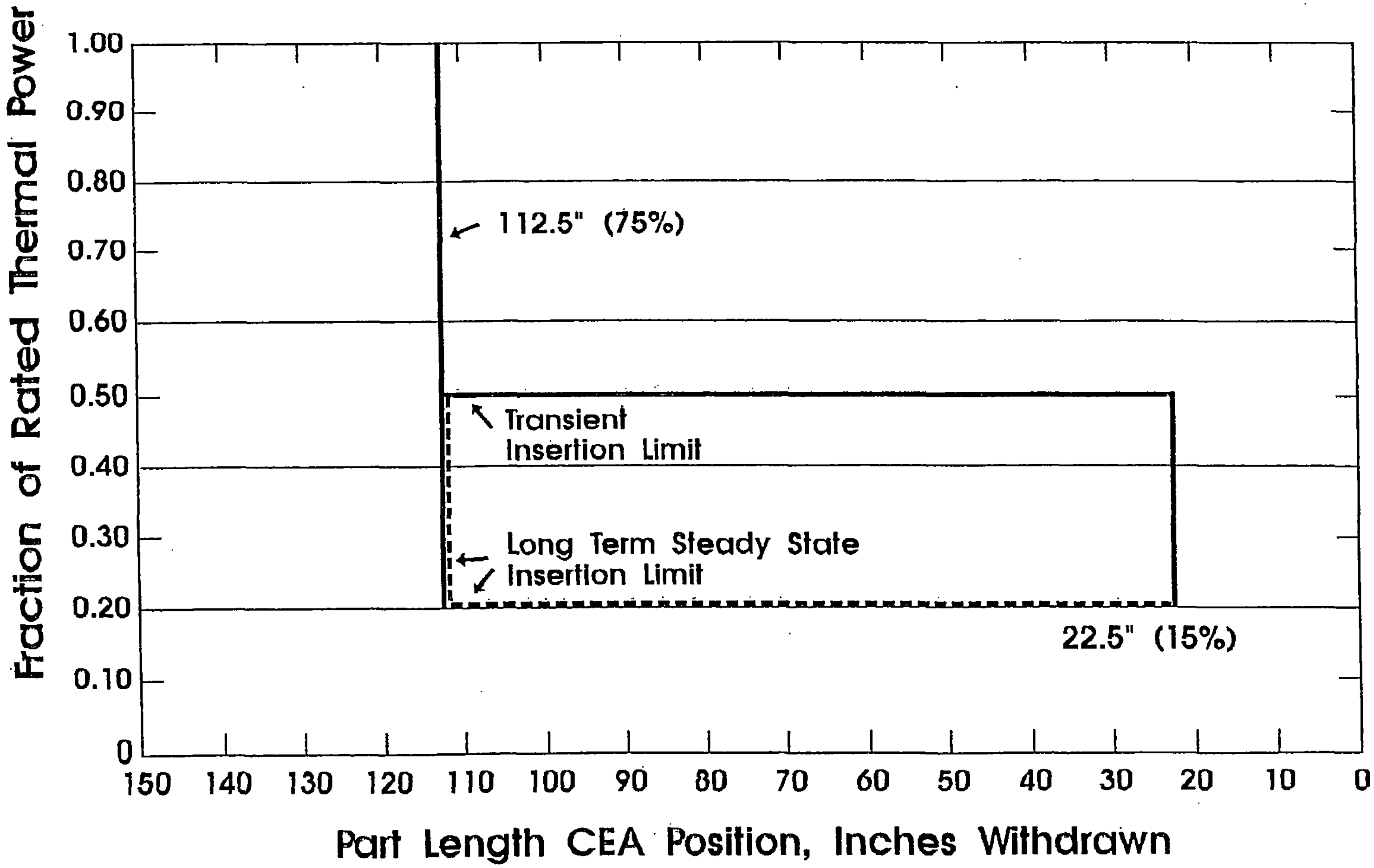


Figure 3.1.103-1

LCS 3.1.103 Part Length CEA Insertion Limits

Bases

The Core Operating Limits Report (COLR) Licensee Controlled Specification (LCS) for Part Length Control Element Assembly (CEA) Insertion Limits provide the part length CEA insertion limits while operating in Mode 1 and reactor power > 20% of RTP. The transient and steady state part length CEA insertion limits are specified graphically as a function of the fraction of rated Thermal Power. The part length CEA limits ensure that safety analysis assumptions for ejected CEA worth and power distribution peaking factors are preserved. Limited deviations from the nominal requirements are permitted with Technical Specification (TS) ACTION statements providing additional compensatory restrictions and time limits. TS Surveillance Requirements provide assurance that necessary system components are OPERABLE and that CEA positions that may approach or exceed acceptable limits are detected, with adequate time for an Operator to take any required Action.

3.1 REACTIVITY CONTROL SYSTEMS

LCS 3.1.105 Control Element Assembly (CEA) Misalignment Power Reduction

All full length CEAs shall be OPERABLE and all full and part length CEAs shall be aligned to within 7 inches of all other CEAs in its group.

VALIDITY STATEMENT: Rev's. 1, 2 and 4 effective 04/16/99, to be implemented within 30 days

APPLICABILITY: MODES 1 and 2 except during PHYSICS TESTS under the Special Test Exemptions of the Technical Specifications.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One non-group 6 full length CEA trippable and misaligned from its group by > 7 inches.	A.1 Initiate THERMAL POWER reduction in accordance with Figure 3.1.105-1 requirements.	In accordance with Figure 3.1.105-1.
B. One group 6 CEA trippable and misaligned from its group by > 7 inches.	B.1 Initiate THERMAL POWER reduction in accordance with Figure 3.1.105-2 requirements.	In accordance with Figure 3.1.105-2.
C. One part length CEA initially ≥ 112.5" misaligned from its group by > 7 inches.	C.1 Initiate THERMAL POWER reduction in accordance with Figure 3.1.105-3 requirements.	In accordance with Figure 3.1.105-3.
D. One part length CEA initially < 112.5" misaligned from its group by > 7 inches.	D.1 Initiate THERMAL POWER reduction in accordance with Figure 3.1.105-4 requirements.	In accordance with Figure 3.1.105-4.

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 Refer to TS 3.1.5.	In accordance with TS 3.1.5.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Refer to LCO 3.1.5	

REQUIRED POWER REDUCTION AFTER SINGLE NON-GROUP 6
FULL LENGTH CEA DEVIATION*

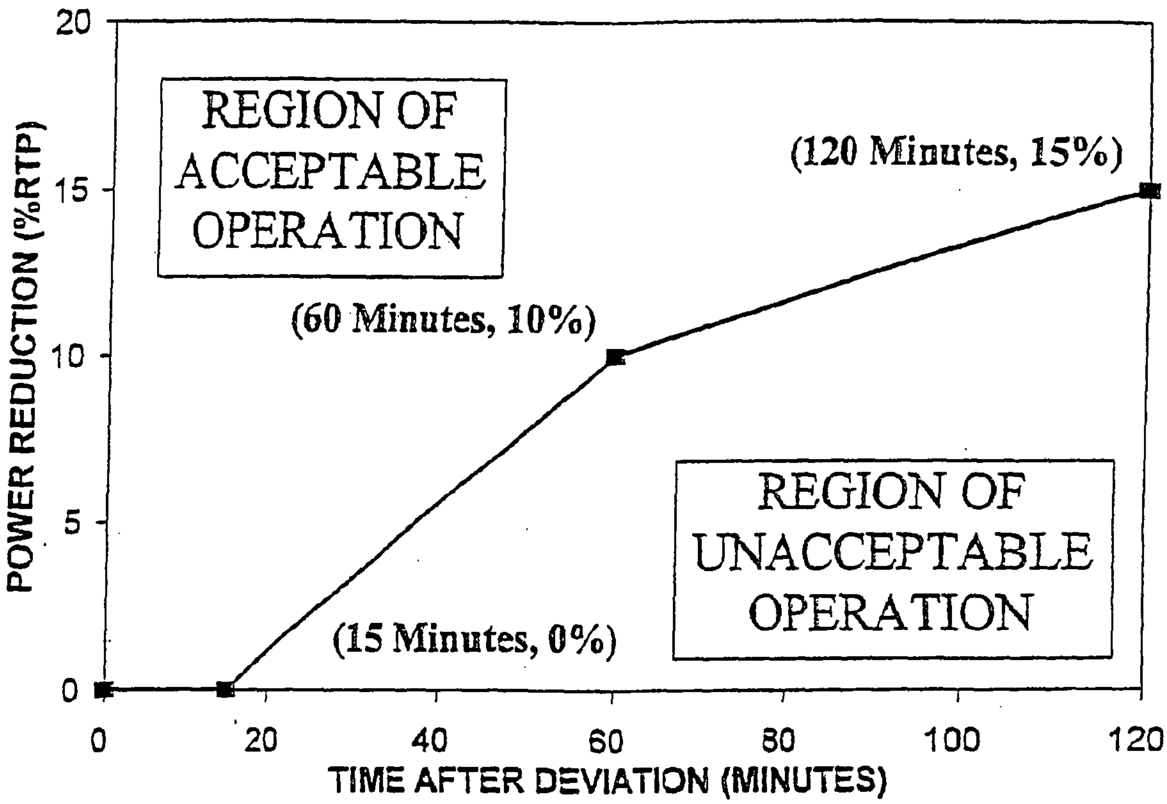


FIGURE 3.1.105-1

* When core power is reduced to 50% of rated power per this limit curve, further reduction is not required by this specification.

REQUIRED POWER REDUCTION AFTER SINGLE GROUP 6
FULL LENGTH CEA DEVIATION*

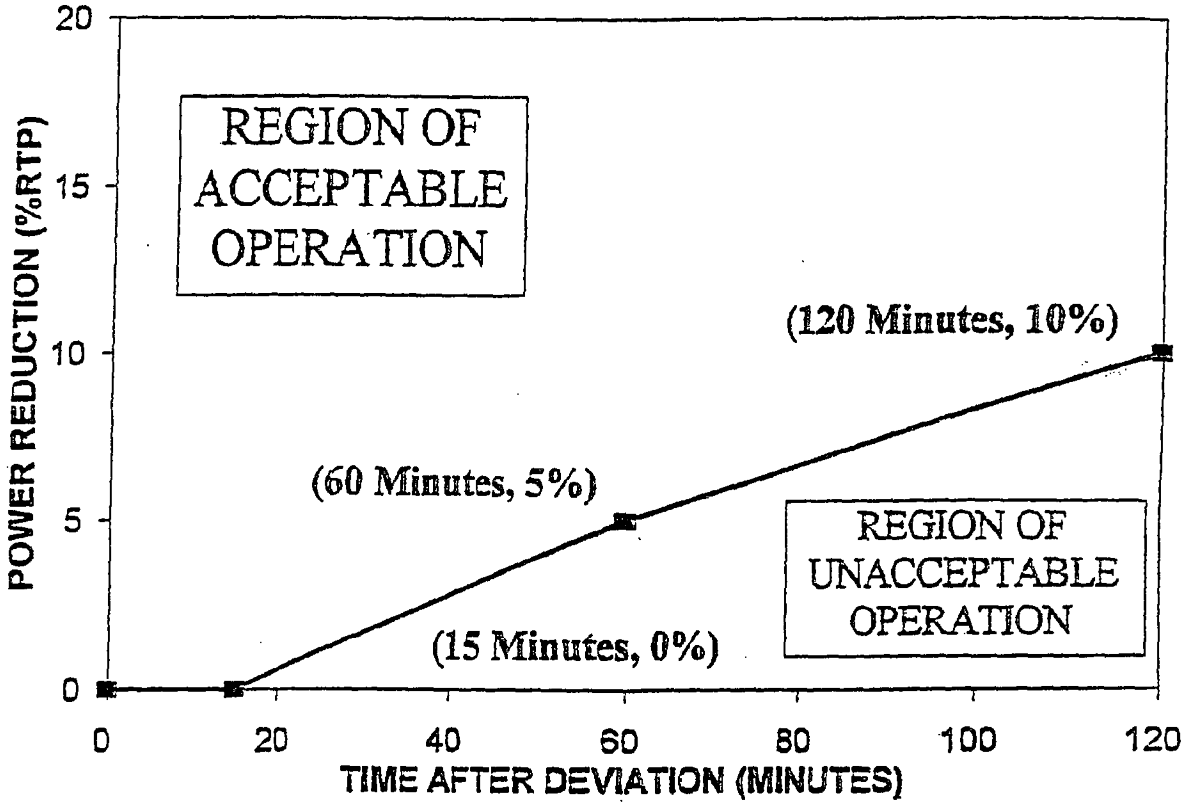
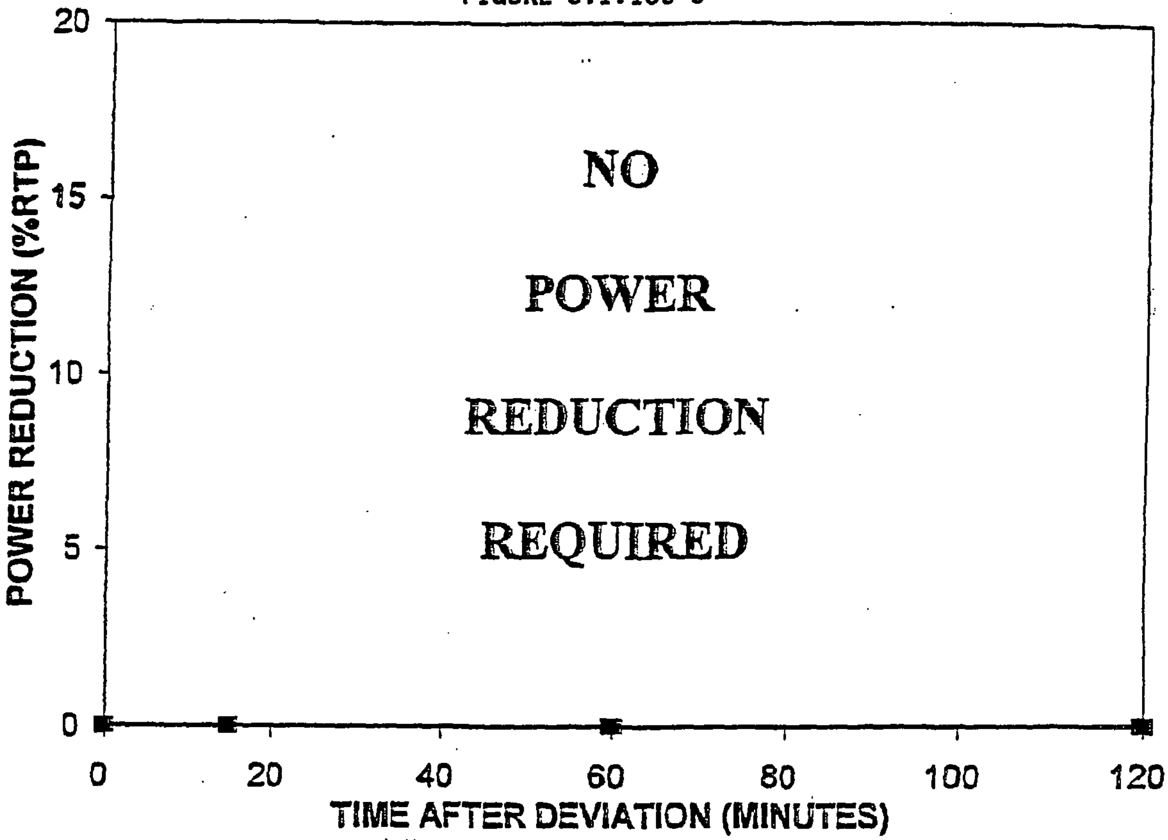


FIGURE 3.1.105-2

* When core power is reduced to 50% of rated power per this limit curve, further reduction is not required by this specification.

REQUIRED POWER REDUCTION AFTER SINGLE PART LENGTH CEA DEVIATION (CEA INITIALLY ≥ 112.5 INCHES WITHDRAWN)

FIGURE 3.1.105-3



REQUIRED POWER REDUCTION AFTER SINGLE
PART LENGTH CEA DEVIATION*
(CEA INITIALLY < 112.5 INCHES WITHDRAWN)

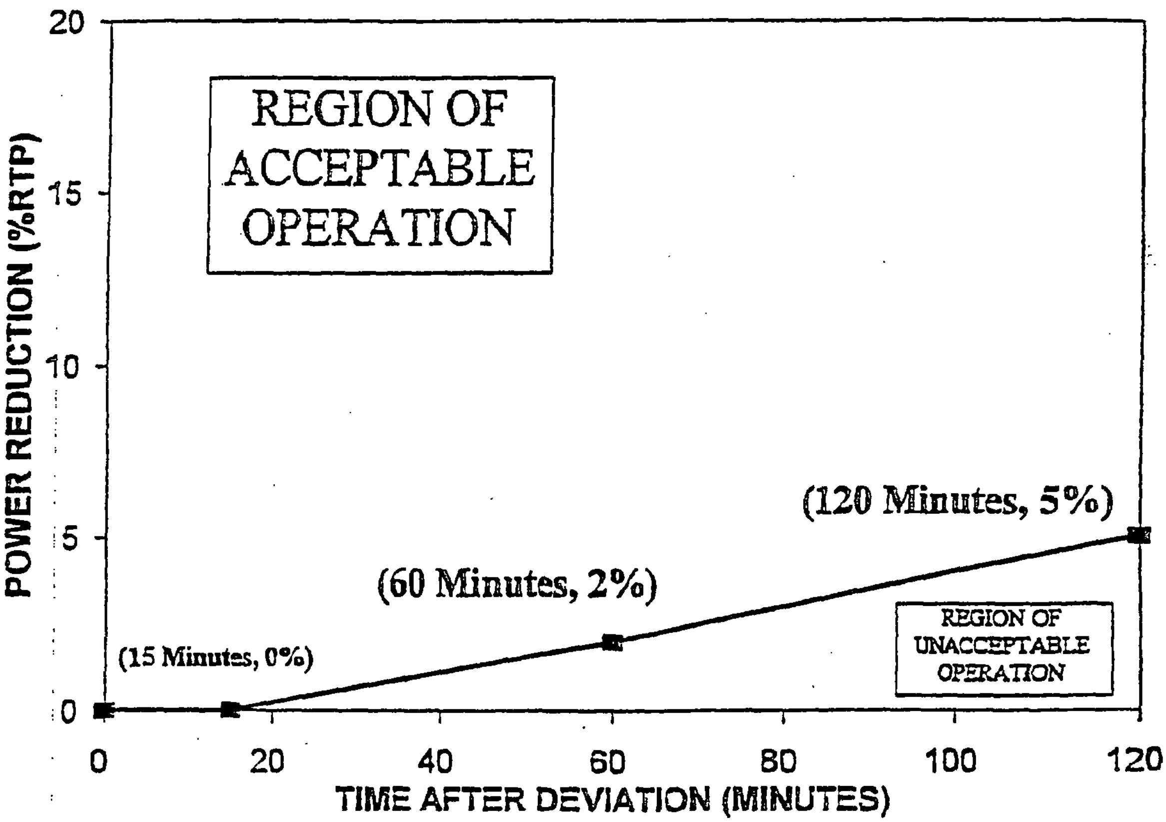


FIGURE 3.1.105-4

* When core power is reduced to 50% of rated power per this limit curve, further reduction is not required by this specification.

LCS 3.1.105 CEA Misalignment Power Reduction

Bases

LCS 3.1.105

The Core Operating Limits Report (COLR) Licensee Controlled Specification (LCS) for Control Element Assembly (CEA) Misalignment Power Reduction provides the power reduction required following a single CEA becoming misaligned from its group by greater than 7 inches while operating in Modes 1 and 2. There are 4 separate power reduction figures provided, with application being dependent on the type of CEA, either "full-length" or "part-length", and the initial position of the "part-length" CEA. For "full-length" CEAs, there are two "sub-types" identified: "non-Group 6" and "Group 6". For "part-length" CEAs, there are two initial conditions identified: "initially ≥ 112.5 inches withdrawn" or "initially < 112.5 inches withdrawn".

The reason for establishing four separate power reduction figures is that full-length group 6 CEAs and/or part-length CEAs are typically used during normal operation. Therefore, a misalignment would most likely involve a CEA in one of these CEA groups. Furthermore, due to the design of the part-length CEAs and their associated insertion limits, it is possible for an inward misalignment to add positive reactivity to the core. Thus, the initial position of a single misaligned part-length CEA must be considered.

The required power reductions are specified graphically as a function of time following the CEA deviation event. For the first 15 minutes, no power reduction is necessary since there is sufficient thermal margin already reserved in the Core Operating Limits Supervisory System (COLSS) or, if COLSS is out-of-service, the amount of thermal margin administratively established by LCS 3.2.101, Departure from Nucleate Boiling Ratio (DNBR). After 15 minutes, a power reduction may be required to increase the thermal margin to offset the build-in of Xenon and its detrimental affect on the radial core power distribution (called "distortion").

Reactor power is required to be reduced to compensate for the increased radial power peaking that occurs following a CEA misalignment. At lower power levels, the potentially adverse consequences of increased radial power peaking can be eliminated.

The magnitudes of the required power reductions differ because of the mechanical design differences between full-length and part-length CEAs and the core physics characteristics due to the fuel load pattern. There are two

(continued)

LCS 3.1.105 CEA Alignment Power Reduction

Bases

major mechanical differences between full-length and part-length CEAs: the lengths and types of neutron absorbers. In a part-length CEA, the neutron absorber is Inconel and is positioned entirely in the lower half of the CEA. In a full-length CEA, there are two types of neutron absorbers: silver-indium-cadmium, located in the bottom 12.5 inches of the CEA, and 136 inches of boron carbide, located above the silver-indium-cadmium.

Since Inconel is neutronically less reactive than boron carbide and silver-indium-cadmium, there will be less of a distortion of the core power distribution as a results of a misalignment of a single part-length CEA initially ≥ 112.5 inches withdrawn. Therefore, the magnitude of the power reduction for a part-length CEA initially ≥ 112.5 inches withdrawn is less than that for a full-length CEA. However, the positive reactivity added by the misalignment of a single part-length CEA initially < 112.5 inches withdrawn and the resulting power increase is more significant than the difference in the absorbers and a power reduction is required to return power to ≤ 50% RTP where there is sufficient margin already reserved.

One of the core physics characteristics established by the fuel load pattern is CEA reactivity. CEA reactivity depends on the power being produced in the fuel assembly into which the CEA is inserted. Analysis of a single group 6 CEA misalignment need only be considered with the power being produced in the fuel assemblies into which a group 6 CEA could be inserted. For all other full-length CEAs, the most adverse conditions must be considered. Due to the physical location of group 6, it is unlikely that misalignment of a single group 6 CEA will be most limiting; and typically it is not. Therefore, the magnitude of the power reduction for a group 6 CEA is less than that for the limiting full-length non-group 6 CEA.

A maximum of 120 minutes is allotted to concurrently reduce power and/or eliminate the misalignment. The 120 minute limit is based solely on the duration evaluated in the applicable analyses. Since there is no safety analysis basis provided beyond the 120 minute limit, Technical Specification 3.1.5 requires that the plant be placed in Mode 3 within 6 hours after reaching the 120 minute limit. However, during the power reduction to achieve Mode 3 conditions, continued efforts to re-align the affected CEA are acceptable and recommended.

At all times throughout a required power reduction, THERMAL POWER shall be reduced by greater than or equal to the amount specified by the appropriate figure for the given time following the CEA deviation.

(continued)

LCS 3.1.105 CEA Misalignment Power Reduction

Bases

The analysis performed to determine the figures contains the following basic assumptions:

1. Only one CEA is misaligned;
2. The magnitude of the required power reduction is determined from the increase in the integrated radial peaking factor(F_r), represented by static and dynamic distortion factors, the Power Operating Limit (POL)-to-F_r ratio and the thermal margin reserved in COLSS as a function of power level;
3. The increase in F_r is evaluated for only 120 minutes;
4. The thermal margin increase accompanying the decrease in core inlet temperature is used to compensate for the thermal margin decrease accompanying the decrease in RCS pressure;
5. The change in the axial power distribution due to the misalignment of a single CEA has been considered, when applicable, in the power reduction curves;
6. Core power is assumed to remain at its initial value for the full-length CEA and the part-length CEA initially ≥ 112.5 inches withdrawn analyses. No credit is taken for the decrease in the power level due to the negative reactivity added as a result of an inward deviation; and
7. The increase in core power for the part-length CEA initially < 112.5 inches withdrawn analysis is explicitly considered.

3.1 REACTIVITY CONTROL SYSTEMS

LCS 3.1.107 SHUTDOWN MARGIN (SDM) – T_{avg} > 200°F

SDM shall be $\geq$ 5.15% $\Delta k/k$.

VALIDITY STATEMENT: Revision 0, effective immediately, to be implemented by December 2, 2005.

APPLICABILITY: Modes 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
Refer to LCO 3.1.1		

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Refer to LCO 3.1.1	

LCS 3.1 REACTIVITY CONTROL SYSTEMS

LCS 3.1.107 SHUTDOWN MARGIN (SDM) – T_{avg} > 200°F

BASES

BACKGROUND	Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. As such, SDM defines the % Δk/k sub-critical that would be obtained immediately following the insertion of all full length control element assemblies (CEAs), assuming the single CEA of highest reactivity worth is fully withdrawn. The SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences assuming the highest reactivity worth CEA remains fully withdrawn. When the CEAs are all verified to be inserted, by both open reactor trip breakers and the CEA position indications, it is not required to assume that the highest reactivity worth CEA is stuck out.
APPLICABLE SAFETY ANALYSES	The minimum required SDM is assumed as an initial condition in the safety analyses. The safety analyses establish an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth CEA stuck out following a reactor trip. When the CEAs are all verified to be inserted, by both open reactor trip breakers and the CEA position indications, it is not required to assume that the highest reactivity worth CEA is stuck out. The acceptance criteria for the SDM are that specified acceptable fuel design limits are maintained. This is done by ensuring that: a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

(continued)

APPLICABLE SAFETY ANALYSES (continued)

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limit AOOs, and ≤ 280 cal/gm energy deposition for the CEA ejection accident).

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements are based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 temperature value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement for MODES 3 and 4 must also protect against:

a. Inadvertent boron dilution;

(continued)

APPLICABLE SAFETY ANALYSES (continued)

b. An uncontrolled CEA withdrawal from a subcritical condition;

c. Startup of an inactive reactor coolant pump (RCP); and

d. CEA ejection.

Each of these is discussed below.

In the boron dilution analysis (Ref. 2), the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life when critical boron concentrations are highest.

The withdrawal of CEAs from subcritical conditions (Ref. 2) adds reactivity to the reactor core, which can cause both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The withdrawal of CEAs also produces a time dependent redistribution of core power.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled CEA withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The startup of an inactive RCP (Ref. 2) will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. An idle RCP cannot, therefore, produce a return to power from the hot standby condition.

(continued)

APPLICABLE SAFETY ANALYSES (continued)	The ejection of a CEA from subcritical conditions (Ref. 2) adds reactivity to the reactor core, which can cause both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a CEA also produces a time dependent redistribution of core power. The SDM satisfies Criterion 2 of the NRC Policy Statement.
LCS	The MSLB (Ref. 2) and the boron dilution (Ref. 2) accidents are the most limiting analyses that establish the SDM requirement. For MSLB accidents, if the LCS is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criterion," limits (Ref. 3). For the boron dilution accident, if the LCS is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable. SDM is a core physics design condition that can be ensured through CEA positioning (regulating and shutdown CEAs) and through the soluble boron concentration.
APPLICABILITY	In MODES 3 and 4, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.7. In MODE 5, SDM is addressed by LCS 3.1.108, "SHUTDOWN MARGIN (SDM)-T_{avg} ≤200°F." In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."
REFERENCES	1. 10 CFR 50, Appendix A, GDC 26. 2. SONGS Units 2 and 3 UFSAR, Section 15 3. 10 CFR 100.

3.1 REACTIVITY CONTROL SYSTEMS

LCS 3.1.108 SHUTDOWN MARGIN (SDM) – T_{avg} ≤ 200°F

SDM shall be ≥ 3.5% Δk/k in Cycle 13.
SDM shall be ≥ 4.0% Δk/k in Cycle 14 and thereafter.

VALIDITY STATEMENT: Revision 1, effective immediately, to be implemented by September 27, 2006.

APPLICABILITY: Mode 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Refer to TS 3.1.2	In accordance with TS 3.1.2.
B. More than one charging pump is functional when the reactor coolant system is at less than full inventory (i.e., pressurizer level < 5%).	B.1 Prevent more than one charging pump from being functional, by verifying that power is removed from the remaining charging pumps.	1 hour
C. Required Action and/or associated Completion Time of Condition B not met.	C.1 Perform a Cause Evaluation.	Within the time specified by the controlling site procedure.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
Refer to LCO 3.1.2 for SDM SR.		
SR 3.1.108.1	Verify no more than one charging pump is functional, by verifying that power is removed from the remaining charging pumps, when the reactor coolant system is at less than full inventory (i.e., pressurizer level < 5%).	Prior to draining the RCS to below 5% pressurizer level, then once per 24 hours.

LCS 3.1 REACTIVITY CONTROL SYSTEMS

LCS 3.1.108 SHUTDOWN MARGIN (SDM) - T_{avg} ≤ 200°F

BASES

BACKGROUND

The reactivity control systems must be redundant and functional of holding the reactor core subcritical when shut down under cold conditions, in accordance with GDC 26 (Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. As such, SDM defines the % $\Delta k/k$ sub-critical that would be obtained immediately following the insertion of all full length control element assemblies (CEAs), assuming the single CEA of highest reactivity worth is fully withdrawn. The SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences assuming the highest reactivity worth CEA remains fully withdrawn. When the CEAs are all verified to be inserted, by both open reactor trip breakers and the CEA position indications, it is not required to assume that the highest reactivity worth CEA is stuck out.

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analyses (Ref. 2) establish an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs with the assumption of the highest worth CEA stuck out following a reactor trip. When the CEAs are all verified to be inserted, by both open reactor trip breakers and the CEA position indications, it is not required to assume that the highest reactivity worth CEA is stuck out. Specifically, for MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that the specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

(continued)

APPLICABLE SAFETY ANALYSES (continued)

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio, fuel centerline temperature limits for AOOs, and $\leq$ 280 cal/gm energy deposition for the CEA ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

An inadvertent boron dilution is defined as a moderate frequency incident (Ref. 2). The core is initially subcritical with all CEAs inserted. A Chemical and Volume Control System malfunction occurs, which causes unborated water to be pumped to the RCS.

The reactivity change rate associated with boron concentration changes due to inadvertent dilution is within the capabilities of operator recognition and control.

The high neutron flux alarm on the startup channel instrumentation will alert the operator to the boron dilution with a minimum of 15 minutes remaining before the core becomes critical.

SDM satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCS

The accident analysis has shown that the required SDM is sufficient to avoid unacceptable consequences to the fuel or RCS as a result of the events addressed above.

The boron dilution (Ref. 2) accident initiated in MODE 5 is the most limiting analysis that establishes the SDM value of the LCS. For the boron dilution accident, if the LCS is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

SDM is a core physics design condition that can be ensured through CEA positioning (regulating and shutdown CEAs) and through soluble boron concentration.

(continued)

APPLICABILITY

In MODE 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.7. In MODES 3 and 4, the SDM requirements are given in LCS 3.1.107, "SHUTDOWN MARGIN (SDM)-T_{avg} > 200°F." In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

ACTIONS

A.1

If SDM is not within limit, refer to TS 3.1.2.

B.1

A Completion Time of 1 hour is adequate for the operator actions necessary to prevent injection from the affected charging pumps.

C.1

If the Condition B, including its Required Action and/or associated Completion Time is not met, a Cause Evaluation should be prepared which will delineate proposed corrective actions. A Cause Evaluation should be prepared within the time specified by the controlling site procedure.

SURVEILLANCE REQUIREMENTS

SR 3.1.108.1

The speed of the boron dilution event is dependent on the rate that the unborated water is injected into the RCS, and on the RCS volume. As RCS volume decreases, the event proceeds more rapidly. By limiting the number of charging pumps that are functional, by verifying that power is removed from the remaining charging pumps when the reactor coolant system is at less than full inventory (i.e., pressurizer level < 5%), the rate of unborated water injection is reduced, so that sufficient time for operator response is ensured.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.108.1

The Frequency of "Prior to draining the RCS to below 5% pressurizer level" ensures that only one charging pump is allowed to be functional prior to entering the condition where this restriction is necessary. The periodic verification frequency of 24 hours provides additional assurance that the charging pump requirement continues to be met as plant conditions change during an outage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. SONGS Units 2 and 3 UFSAR, Section 15

3.2 POWER DISTRIBUTION LIMITS

LCS 3.2.100 Linear Heat Rate (LHR)

LHR shall not exceed 12.8 kW/ft.

VALIDITY STATEMENT: Effective 30 days after approval.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
	Refer to LCO 3.2.1	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Refer to LCO 3.2.1	

LCS 3.2.100 Linear Heat Rate (LHR)

BASES

The COLR limitation on LHR ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F. Actions and Surveillance Requirements are provided by the Technical Specifications (TS).

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory system (COLSS) or the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the LHR does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate. With the reactor operating at or below this calculated power level the LHR limit is not exceeded.

The COLSS calculated core power and the COLSS calculated core power operating limits based on LHR are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the LHR operating limit for normal steady state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the LHR includes appropriate penalty factors which provide, with a 95/95 probability/ confidence level, that the maximum LHR calculated by COLSS is conservative with respect to the actual maximum LHR existing in the core. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering design factors, axial densification, software algorithm modelling, computer processing, rod bow and core power measurement.

The core power distribution and a corresponding power operating limit based on LHR are more accurately determined by the COLSS using the incore detector system. The CPCs determine LHR less accurately with the excore detectors. Therefore, when COLSS is not available the TS LCOs are more restrictive due to the uncertainty of the CPCs. However, when COLSS initially becomes inoperable, the added margin associated with CPC uncertainty is not immediately required and a 4 hour Action is provided for appropriate corrective action.

Parameters required to maintain the operating limit power level based on LHR, margin to DNB and total core power are also monitored by the CPCs assuming minimum core power of 20% RATED THERMAL POWER. The 20% Rated Thermal Power

(continued)

BASES (continued)

threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Therefore, in the event that the COLSS is not being used, operation within the DNBR limits with COLSS out of service can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty penalty factors plus those associated with startup test acceptance criteria are also included in the CPCs.

While operating with the COLSS out of service, the CPC calculated LHR is monitored every 15 minutes to identify any adverse trend in thermal margin. The increased monitoring of LHR during the 4 hour action period ensures that adequate safety margin is maintained for anticipated operational occurrences and no postulated accident results in consequences more severe than those described in Chapter 15 of the UFSAR.

3.2 POWER DISTRIBUTION LIMITS

LCS 3.2.101 The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained by one of the following methods:

a. Maintaining Core Operating Limit Supervisory System (COLSS) calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR (when COLSS is in service, and either one or both control element assembly calculators (CEACs) are OPERABLE);

b. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by 13.0% RTP (when COLSS is in service and neither CEAC is OPERABLE);

c. Operating within limits as specified in Figure 3.2.101-1A for initial power ≥ 90% RTP or Figure 3.2.101-1B for initial power < 90% RTP using any OPERABLE core protection calculator (CPC) channel (when COLSS is out of service and either one or both CEACs are OPERABLE); or

d. Operating within limits as specified in Figure 3.2.101-2 using any OPERABLE CPC channel (when COLSS is out of service and neither CEAC is OPERABLE).

VALIDITY STATEMENT: Rev. 2 effective 4/16/99, to be implemented within 30 days.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
	Refer to LCO 3.2.4	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Refer to LCO 3.2.4	

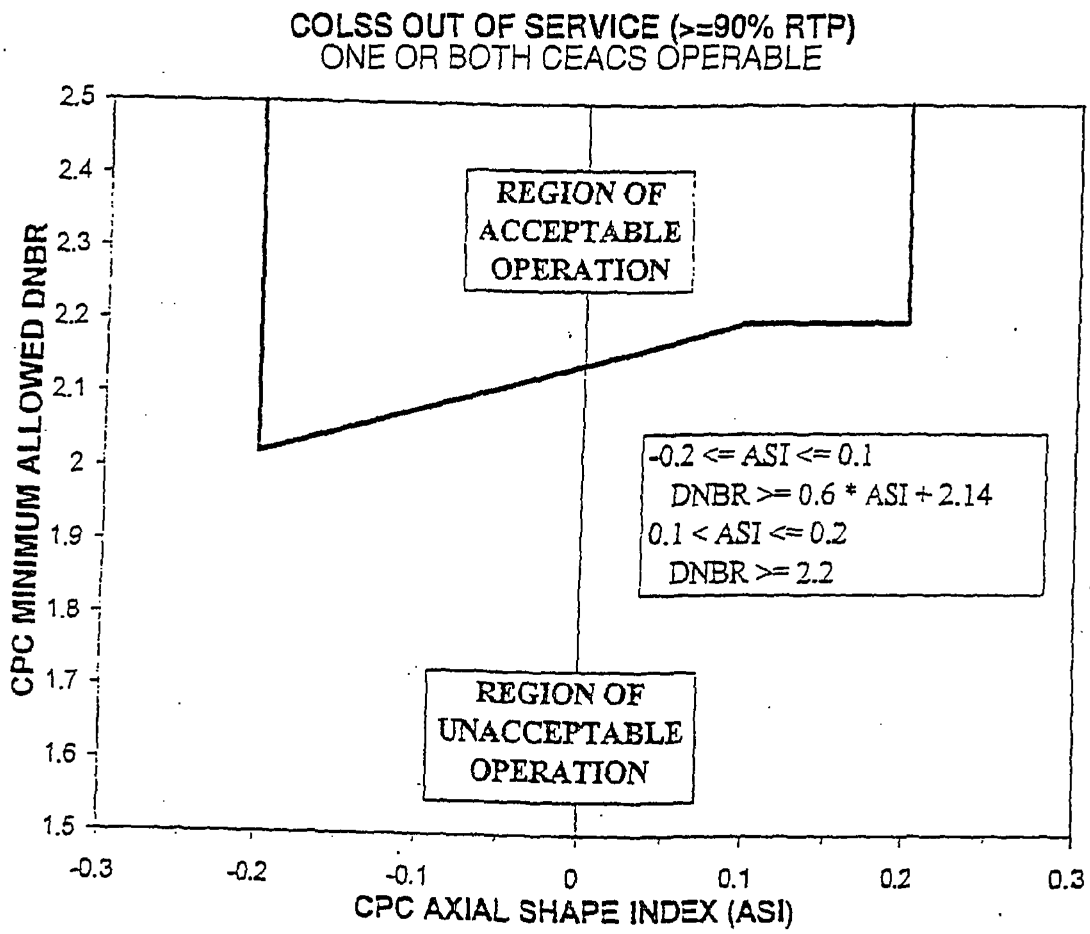


Figure 3.2.101-1A DNBR OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS
- COLSS OUT OF SERVICE
- ONE OR BOTH CEACS OPERABLE

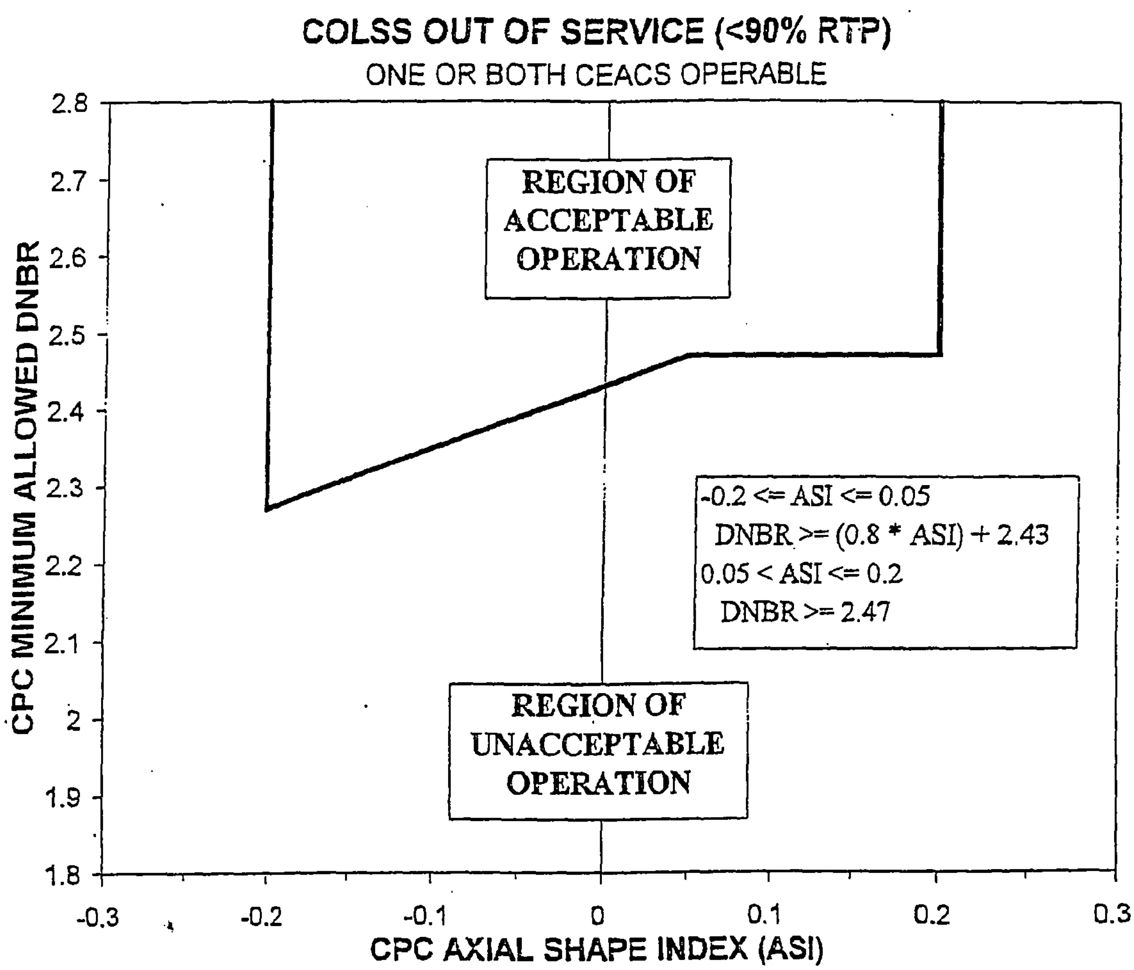


Figure 3.2.101-1B DNBR OPERATING LIMIT BASED ON CORE PROTECTION CALCULATOR
- COLSS OUT OF SERVICE
- ONE OR BOTH CEACS OPERABLE

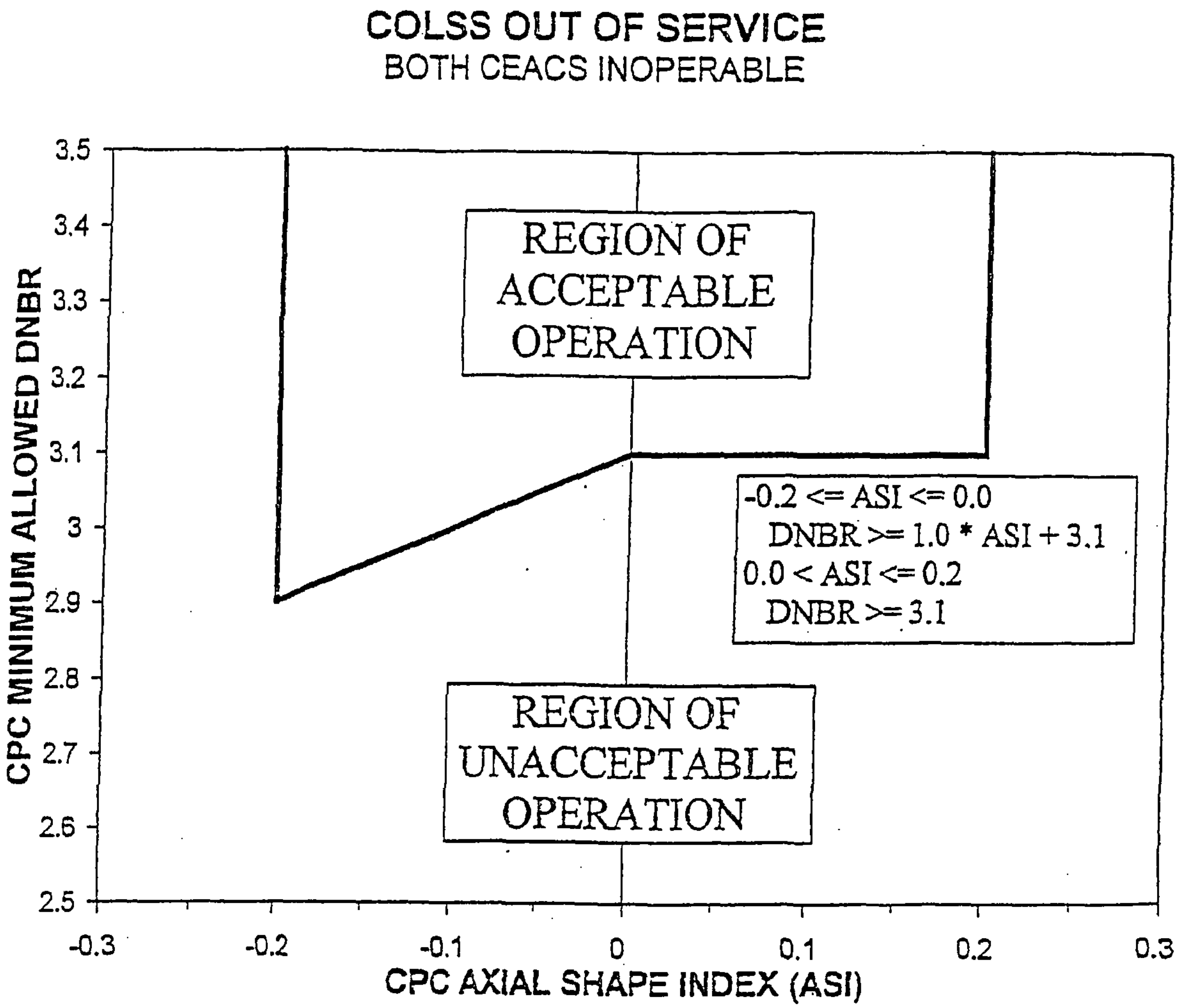


Figure 3.2.101-2 DNBR OPERATING LIMIT BASED ON CORE PROTECTION CALCULATOR
- COLSS OUT OF SERVICE
- BOTH CEACS INOPERABLE

LCS 3.2.101 DNBR

BASES

The COLR limitation on DNBR as a function of Axial Shape Index (ASI) represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient. The TS provides the required Actions and Surveillance Requirements to ensure that the minimum DNBR is maintained.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) or the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate COLR specified limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating power limit corresponding to the allowable minimum DNBR. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limit calculated by COLSS (based on the minimum DNBR limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering design factors, state parameter measurement, software algorithm modeling, computer processing, rod bow and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. In the event that the COLSS is not being used, the DNBR margin can be maintained by monitoring with any OPERABLE CPC channel so that the DNBR remains above the predetermined limit as a function of Axial Shape Index. The above listed uncertainty penalty factors are also included in the CPCs, which assume a minimum of 20% of RATED THERMAL POWER. For the condition in which one or both CEACs are operable, the thermal margin requirements are given as a function of power level. One requirement applies to ≥ 90 % RTP and the other applies to < 90% RTP. The 20% RATED THERMAL POWER threshold is due to the excore neutron flux detector system being less accurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The additional CPC uncertainty terms for transient protection are removed from the COLR figures since the curves are intended to monitor the LCO only during steady state operation.

The core power distribution and a corresponding POL based on DNBR are more accurately determined by the COLSS using the incore detector system. The CPCs

(continued)

determine DNBR less accurately using the excore detectors. When COLSS is not available the TS LCOs are more restrictive due to the uncertainty of the CPCs. However, when COLSS initially becomes inoperable the added margin associated with CPC uncertainty is not immediately required and a 4 hour ACTION is provided for appropriate corrective action.

A DNBR penalty factor has been included in the COLSS and CPC DNBR calculation to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. In design calculations, the penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

While operating with the COLSS out of service, the CPC calculated DNBR is monitored every 15 minutes to identify any adverse trend in thermal margin. The increased monitoring of DNBR during the 4 hour action period ensures that adequate safety margin is maintained for anticipated operational occurrences and no postulated accident results in consequences more severe than those described in chapter 15 of the UFSAR.

3.2 POWER DISTRIBUTION LIMITS

LCS 3.2.102 Core average Axial Shape Index (ASI) shall be within the following limits:

a. COLSS OPERABLE $-0.27 \leq ASI \leq +0.27$

b. COLSS OUT OF SERVICE $-0.20 \leq ASI \leq +0.20$

VALIDITY STATEMENT: Revisions 2 and 3 effective 4/16/99, to be implemented within 30 days.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP except during PHYSICS TESTS under the Special Test Exemptions of the Technical Specifications.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
	Refer to LCO 3.2.5	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Refer to LCO 3.2.5	

B LCS 3.2.102 ASI

BASES

The Axial Shape Index (ASI) is a measure of the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers. This specification is provided to ensure that the core average ASI is maintained within the range of values assumed as an initial condition in the safety analyses.

The ASI can be determined by utilizing either the Core Operating Limit Supervisory System (COLSS) or any OPERABLE Core Protection Calculator (CPC) channel. The real time monitoring capability and accuracy of COLSS allows COLSS to monitor power limit margins closely. Consequently, the ASI limit is broader than it would be with the same core without COLSS. The COLSS continuously calculates the ASI and compares the calculated value to the parameter established for the COLSS ASI alarm limit. In addition, there is an uncertainty associated with the COLSS calculated ASI; therefore the COLSS ASI alarm limit includes this uncertainty. If the LCO is exceeded, COLSS alarms are initiated. The ASI limit is selected so that no safety limit will be exceeded as a result of an anticipated operational occurrence, and so that the consequence of a design basis accident will be acceptable.

3.4 REACTOR COOLANT SYSTEM (RCS)

LCS 3.4.100 RCS DNB (Pressure, Temperature, and Flow) Limits

RCS Parameters for pressurizer pressure, cold leg temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ 2025 psia and ≤ 2275 psia;

b. RCS cold leg temperature (Tc):
   1. For THERMAL POWER less than or equal to 30% RTP, 522°F ≤ Tc ≤ 558°F ,
   2. For THERMAL POWER greater than 30% RTP, 535°F ≤ Tc ≤ 558°F;

c. RCS total flow rate ≥ 396,000 gpm.

VALIDITY STATEMENT: Revisions 0, effective 03/03/09, to be implemented within 30 days.

APPLICABILITY: Modes 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
	Refer to LCO 3.4.1	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Refer to LCO 3.4.1	

BASES

Refer to Technical Specification 3.4.1 Bases

3.9 REFUELING OPERATIONS

LCS 3.9.100 Boron Concentration Limit

With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of following reactivity conditions is met.

a. $K_{eff} \leq 0.95$, or

b. Boron concentration $\geq$ 2600 ppm.

VALIDITY STATEMENT: Rev. 3 effective 11/18/05, to be implemented by 12/2/05.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The more restrictive of the following not met: a. $K_{eff} \leq 0.95$, or b. Boron concentration $\geq$ 2600 ppm.	A.1 Suspend all operations involving CORE ALTERATIONS or positive reactivity changes. AND A.2 Initiate and continue boration at $\geq$ 40 gpm of a solution containing adequate boron concentration until K_{eff} is reduced to $\leq$ 0.95.	Immediately Immediately
B. More than one charging pump is functional.	B.1 Prevent more than one charging pump from being functional by verifying that power is removed from the remaining charging pumps.	1 hour
C. Required Action(s) and/or associated Completion Time of Condition A not met.	C.1 Perform a Cause Evaluation.	Within the time specified by the controlling site procedure.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.100.1	The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis.	72 hours
SR 3.9.100.2	Verify that no more than one charging pump is functional, by verifying that power is removed from the remaining charging pumps.	24 hours

LCS 3.9.100 Boron Concentration Limit

BASES

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The accident analysis also assumes that only a single charging pump injects during the boron dilution event. The value of 0.95 or less for K_{eff} includes a conservative allowance for uncertainties.

If the Condition A or B, including its Required Action(s) and/or associated Completion Time is not met, a Cause Evaluation should be prepared which will delineate proposed corrective actions. A Cause Evaluation should be prepared within the time specified by the controlling site procedure.

5.0 ADMINISTRATIVE CONTROLS

LCS 5.0.105 Core Operating Limits Report (COLR) Analytical Methods

VALIDITY STATEMENT: Rev. 3 effective 12/15/09, to be implemented within 60 days.

5.0.105.1 The following Technical Specification 5.7.1.5 analytical methods (identified by report number, title, revision, date, and any supplements), previously reviewed and approved by the NRC, shall be used to determine the core operating limits. Changes to the analytical methods are controlled in accordance with 10CFR50.59.

1a. CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974.

1b. CENPD-132P, Supplement 1, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," February 1975.

1c. CENPD-132-P, Supplement 2-P, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," July 1975.

1d. CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS," June 1985.

1e. CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.

2a. CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model, August 1974.

2b. CENPD-137, Supplement1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model, January 1977.

2c. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB C-E Small Break LOCA Evaluation Model, April 1998.

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5.0.105.1 (continued)

3. CEN-356(V)-P-A, Revision 01-P-A, "Modified Statistical Combination of Uncertainties," May 1988.

4. CEN-635(S), "Identification of NRC Safety Evaluation Report Limitations and/or Constraints on Reload Analysis Methodology," Rev. 00, February 1999.

5. SCE-9801-P-A, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3," June 1999.

6. Letter, dated May 16, 1986, G. W. Knighton (NRC) to K. P. Baskin (SCE), "Issuance of Amendment No. 47 to Facility Operating License NPF-10 and Amendment No. 36 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 3 SER).

7. Letter, dated January 9, 1985, G. W. Knighton (NRC) to K. P. Baskin, "Issuance of Amendment No. 30 to Facility Operating License NPF-10 and Amendment No. 19 to Facility Operating License NPF-15," San Onofre Nuclear Generating Station Units 2 and 3 (Cycle 2 SER).

8. CENPD-404-P-A, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001.

9. SCE-0901-A, "PWR Reactor Physics Methodology Using Studsvik Design Codes," Rev 0, December 2009.

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